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A REPORT ON THE SAFETY ASPECTS
OF THE HOMOGENEOUS REACTOR EXPERIMENT

OAK RIDGE NATIONAL LABORATORY
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CARBIDE AND CARBON CHEMICALS DIVISION
UNION CARBIDE AND CARBON CORPORATION

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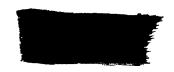OAK RIDGE, TENN

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by

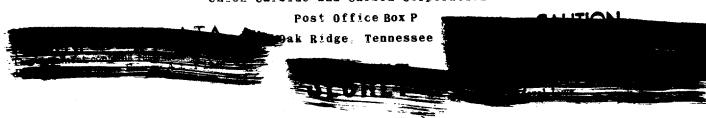Members of the Reactor Technology
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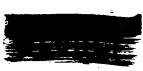20.   Hymer L. Friedell, Western Reserve University

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The Homogeneous Reactor Experiment, a 1000 kw, high pressure $UO_2SO_4 \cdot H_2O$ circulating chain reactor, presents no hazard unless the radioactive soup escapes from the pressurized system in large quantity. All other hazards, such as those connected with routine waste disposal, etc., are of essentially negligible importance.

According to Reactor Safeguard Committee formulas a large scale accident does not lead to a catastrophic area contamination or loss of life. Either ample time is available to evacuate inhabited sites near the reactor or the cloud passes over in a sufficiently short time to limit the dosage to reasonable values. The following table gives calculated radioactive hazards, assuming 100% of the activity from 1000 kw operation is vaporized:

	Wind velocity (mi/hr)	0.1	2.0	4.0
At X-10 (0.5 miles)	Accumulated Roentgen	179	16.2	9.3
	Time from site	5 hrs	1/4 hr	1/8 hr
2 miles from site	Accumulated Roentgen	34.0	3.1	1.8
	Time from site	20 hrs	1 hr	1/2 hr
At K-25 or Y-12 (5 miles)	Accumulated Roentgen	11.3	1.0	0.6
	Time from site	50 hrs	2.5 hrs	1.25 hrs
At Oak Ridge (8 miles)	Accumulated Roentgen	6.4	0.6	0.3
	Time from site	80 hrs	4 hrs	2 hrs

There are four conceivable accident sequences which might cause the pressure container system to rupture and release the radioactive soup. These are:

(a) A mechanical failure in some part of the pressure system during normal operation. The possibility of this is vanishingly small since all pressure parts leading to the atmosphere are designed with a safety factor of at least 10.

(b) Very large sudden reactivity excursions. Instantaneous reactivity excursions of less than 2% can be withstood by the system without damage. Reactivity excursions greater than 2%, and up to some unknown value, would cause the reactor core tank to rupture, violently discharging the soup into the reflector space, whereupon, the system becomes sub-critical by substantial amounts. Activity is not released to the atmosphere because of the stronger outer steel shell. The k excursion required to rupture the core tank sufficiently violently so that the outer steam shell is also ruptured is not known. Qualitatively it should be rather large. A prompt k jump of the order of 2% is believed to be the largest k jump which any reactor now contemplated or in existence can withstand without ill effect. This reactor can withstand this huge prompt k rise because of its enormous negative temperature coefficient, 1/9% Δk per °C.

(c) Aerial bombing or sabotage of the HRE. As seen from the table in the preceding paragraph, even this would probably not cause catastrophic conditions.

(d) A H_2-O_2 explosion of recombining gases. While these recombining systems have not been completely designed, favorable experience in the heavy water manufacturing plants and with the Canadian reactor leads us to the conclusion that, provided proper precautions are taken, the hazard from an explosion in this system of sufficient violence to cause a rupture of the high pressure soup system is rather small.

II. THE REACTOR AND ITS OPERATIONS

DESCRIPTION OF THE REACTOR

The proposed Homogeneous Reactor Experiment (HRE) consists of approximately 2 kg of uranyl sulfate in ordinary water solution. The reacting solution is pumped through the reactor vessel and heat exchanger. The reactor vessel contains about 50 liters and operates at a pressure of 1000 pounds per square inch and a temperature of approximately 250°C. This high temperature and pressure are sufficient to keep the bubble volume manageably small and give an overall power efficiency of 15%. Given 1000 kw of heat output of the reactor, the useful power will be of the order of 150 kw (net). The reactor is surrounded by a heavy water reflector enclosed in a pressure shell, and is also maintained at a pressure of 1000 pounds per square inch.

Reactivity of this device may be controlled by changing solution concentration, moving absorbing rods, and adjusting reflector level. This reactor has a very high negative temperature coefficient (1/9% k/°C) and is, therefore, inherently safe to operate.

The heat exchanger through which the reacting solution circulates is a liquid-liquid, shell and tube type exchanger. Steam is generated on the shell side of the exchanger and is used to run a turbine. The steam pressure can be varied by a throttling valve to regulate the power removed from the reactor. It is planned to run at steam pressure varying from 200 psi at 1000 kw to 450 psi at 200 kw.

The solution is degassed by centrifugal separation in the reactor. The gases are then collected and recombined outside of the reactor core.

The heavy water reflector is circulated and the temperature in the reflector system is regulated by a heat exchanger, and it is possible to dump the reflector through a valve to ever-safe dump tanks. The reflector pressure shell, of a minimum thickness of 3 in., will be a steel sphere 39 in. I.D., with a 24 in. I.D. opening in the top so that the reactor tank may be withdrawn. Several thimbles which protrude into the reflector through the shell will provide space for nuclear instrumentation.

The uranyl sulphate solution system will include ever-safe dump tanks which provide storage space for solution removal from the reactor, means for changing soup concentration by evaporating or adding water to the solution. A low capacity pump will feed the solution to the reactor.

A complete detailed description of the reactor may be found in ORNL 730, *Homogeneous Reactor Experiment Feasibility Report*.[1]

PROPOSED SITE FOR CONSTRUCTION

It is recommended that the Homogeneous Reactor Experiment be located at a site 0.6 miles southeast of the center of the present ORNL area (see Fig. 1). This places the reactor in the center of an uninhabited valley approximately 4 miles long and 0.5 miles wide. The general orientation of the axis of the valley in N.E. and S.W. and it is parallel to Bethel and other neighboring valleys. The Clinch River terminates both ends of the valley, as well as forming a natural boundary of approximately 2-mile radius for a 180° swing from the site.

Between the recommended site at elevation 820 feet, and Bethel Valley, containing ORNL at approximately 820 feet elevation, is Haw Ridge averaging about 980 feet in elevation. This Ridge is continuous except for a short break, known as White Oak Creek gap (770 feet), through which a portion of Bethel Valley drainage occurs. This gap is about 0.4 miles west of the site. and in view of the local weather conditions is not expected to represent any significant decrease in the natural protection afforded to ORNL by the intervening ridge. To the south and east of the site is a large area of high and rough terrain extending south to the Clinch River.

Within a radius of 1.9 miles all of the land is owned by the AEC and is already a security controlled (and patrolled) area. Within a 2-mile radius approximately 10 acres of farm land is not owned or controlled. A radius of 2.2 miles does not substantially increase this privately owned and uncontrolled area.

NORMAL OPERATING PLAN

General Operating Characteristics The following methods of changing k are available to the operator:

1. Change soup concentration.

2. Change reflector level.

3. Moving safety plates and regulating plate.

For normal operations including startup and shutdown, it is proposed to use soup concentration as the shim and the "regulating" rod for fine adjustment. While in operation the safety plates are entirely withdrawn and ready for action, and the reflector level adjusted so that the core is just covered. In order to maintain the effectiveness of the safety system, at no time will the reactor be operated with less than half-full reflector, and interlocks are provided to prevent this occurrence. With the limited amount of uranium provided the operator, it will be impossible, according to static criticality calculations, for the operation to go critical at less than a half-full reflector if the interlocks fail.

For normal operation we may assume that the following information has been attained in previous critical experiments: critical mass as a function of temperature, reflector level, circulation rate; effects of neutron absorbing control rods; temperature coefficient of reactivity of the reactor; and inherent operating stability of the reactor. Two fundamental characteristics of the reactor should be pointed out.

1. Because of the very large temperature coefficient of reactivity, there is a strong coupling between the temperature and reactivity. In equilibrium the average operating temperature of the soup will be determined by the multiplication constant of the reactor, and the operating temperature is determined by the control plate position or other reactivity control elements.

2. Again because of the big temperature coefficient of reactivity, the reactor power is determined by the power demand, *i.e.* by the amount of steam that is drawn off of the soup heat exchanger (boiler). This means that the turbine governor, which regulates steam flow to the turbine, regulates the power level of the reactor. Also this means that when the reactor circulation—and therefore cooling—falls to that taken out by convection to reflector, the power of the reactor also drops to that level.

Start-Up Procedures. Before attempting to start up the reactor, it is necessary to check the conditions of the control elements of the reactor. The regulating and safety plates are in their "out" position. The reflector liquid is in the reflector dump tank at approximately 20°C. Soup solution is in the soup dump tank at approximately 100°C. The soup concentration should be approximately 20 grams per liter, which gives a mass about 90% of the critical for the core surrounded by a full cold reflector. The soup should be concentrated or diluted to the above value if not at that concentration.

It is necessary to make a preliminary check of the instrumentation in order to ascertain any failures or doubtful conditions of operation. This is especially important in regard to the safety mechanisms of the reactor. This instrumentation is described in detail in ORNL 730, *Homogeneous Reactor Experiment Feasibility Report* [1]

Nuclear instrumentation is quite similar to that used on the Materials Testing Reactor and has been described and discussed previously. [6][11] The essential instruments for operation are two boron coated ionization chambers for flux level measurement, one differential ionization chamber with a logarithmic differentiator to indicate period, two fission chambers to measure low level flux, and indicating instruments for soup pressure and temperature, steam pressure, and soup concentration. The flux level and period instrument circuits are directly connected to the safety rod trip circuit. This part of the safety mechanism is essentially the same as that used in the Materials Testing Reactor.

The start-up procedure is outlined below. The steps before attaining criticality are performed cautiously, and increasing soup concentration and power load on the reactor is done slowly.

1. Introduce a neutron source if residual (γ-n) background is not high enough, withdraw safety plates.

2. Fill core with 100°C soup (concentration ~90% of critical concentration with full reflector) and circulate through entire soup system by feed pump.

3. Cautiously fill reflector to operating level and heat to ~100°C.

4. Simultaneously pressurize to ~1000 psi the soup and reflector.

5. Raise soup temperature to ~200°C with building steam supplied to the boiler.

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6. Attain the critical condition by using soup concentrator and "regulating plate" ($\sim 1\%$ Δk); concentration increased until $\sim \frac{1}{2}\%$ Δk from critical, then withdraw regulating plate until reactor becomes critical.

7. Adjust regulating plate to about the midpoint of its effective stroke.

8. Increase soup temperature to $\sim 235\,^\circ$C by slowly increasing soup concentration and using regulating plate if necessary (temperature increased from $200\,^\circ$C by reactor power).

9. Warm up turbine with reactor steam and bring up to speed (~ 100 kw power from reactor, no electrical load on generator).

10. Increase load on generator to that equivalent to 1000 kw of heat from the reactor. The turbine governor increases the steam flow and the reactor power comes up to 1000 kw level as steam flow increases.

11. Shut down reactor.

    a. Shut off steam to turbine;

    b. Decrease soup concentration (soup temperature will slowly decrease to steam temperature $\sim 200\,^\circ$C, reactor will go subcritical with further dilution and soup and boiler water temperature will fall gradually, due to large mass of boiler water);

    c. Continue soup dilution until reactor is subcritical at a low temperature (<20 gm/liter).

PLAN OF EXPERIMENTS WHICH WILL BE PERFORMED ON THE REACTOR,
INCLUDING THE LIMITATIONS PLACED ON EXPERIMENTS

The Homogeneous Reactor Experiment is intended to serve as a pilot model for future reactors of this type. Since it is not designed to serve as a neutron source for experimental work, or for radiation testing of foreign materials, or for the production of isotopes. no experimental holes are provided. The only means of introducing or removing an absorbing material adjacent to the core will be by the control rods and the ionization chamber instrument thimbles. The control rods will have a rapid means of introducing and very slow means of removing absorbing material. The ionization chambers will have a low, constant absorption.

The experiments on the reactor itself have not been planned in detail. In general, they will be for the purpose of checking actual operation with predicted operation, the most important of which are

Heat transfer coefficients

Temperature coefficient of reactivity

Power (gas) coefficient of reactivity

Effect of variables on gas evolution such as temperature pressure, and probably catalysts

Comparison of reflector level vs soup concentration as a method of shimming

Effects of temperature and time upon chemical and radiation stability of soup

Making a heat balance on the system

THE NORMAL SCHEDULE OF CHEMICAL PROCESSING AND
DISPOSAL OF REACTOR PRODUCTS

A normal schedule of chemical processing in the HRE is not believed to be necessary. Long-term (several years) operation is not contemplated, and since the rate of depletion of fissionable material is rather low, the fission product buildup rate will also be rather low. Whenever the experiment is completed, or should it ever be necessary to re-process the solution for any reason, such could easily be accomplished in the ORNL separation pilot plants using the "Purex" process. The active wastes from this operation would be processed and stored in the existing ORNL tank farm and waste disposal system.

THE NORMAL METHODS OF DISPOSAL OF AQUEOUS OR OTHER SOLUTIONS

Almost all of the heat of the reactor will be removed in a manner that will not cause any induced activity in the cooling water. One small stream of about 5 gpm will flow over the heavy steel head, the heavy flange and upper part of the pressure shell of the reactor. The area to be cooled is small, the film is quite thin, and the neutron flux is low ($\sim 10^{10}$). After a ten minute decay period the induced activity level will be below six millicuries per day (or 2×10^{-4} microcuries per cubic centimeter of water, mostly due to Na and Cl) which is below tolerance and may be safely added to the other cooling water and discharged to the small creek flowing past the site to the south. This small creek joins Melton Branch of White Oak Creek which in turn is constantly monitored and discharged to the Clinch River with the balance of all ORNL active wastes. Should further experience indicate that the activity will not be below tolerance, it is not very difficult or expensive to install a small catch tank cooler and recycle the same water, in which case no aqueous activities would be discharged. Appendix I gives the solid contents of the water and the amount of activity induced therein.

DESCRIPTION OF SAFETY MECHANISM OF THE REACTOR

1. Mechanisms Available for Shutting Off the Nuclear Reaction.

A. *Safety Plates* - containing about 9% k when reflector is full; release time about 10 milliseconds; 4 g initial acceleration; time to be fully effective about 0.2 second.

B. *Reflector Dump* - about 40% k; release time about 1 second; time for complete emptying about 10 seconds.

C. *Soup Dump* - release time about 1 second; completely emptied in 2 minutes.

D. *Cessation of Soup Circulation* - if circulating pump stops chain reaction will stop because soup temperature increases. It is estimated that from the time that the pump power is cut off to the time the fluid flow falls to about 10% of normal is about 1 second.

In addition, the reactor has a "built in" safety system in that its temperature coefficient of reactivity is enormous—1/9% δk per °C at 250°C. The stabilizing effect of the large temperature coefficient is described in some detail in the accompanying feasibility report, ORNL 730.

The safety plates are suspended by electromagnets. The 4 g initial acceleration is provided by springs. A number of signals will cause the magnets to release as follows:

A. Excessive flux levels as determined by ion chambers;

B. Flux increases at periods of less than 1 second;

C. Excessive core temperature;

D. A large number of other signals indicative of misoperation or component failures;

E. Signal from the operator.
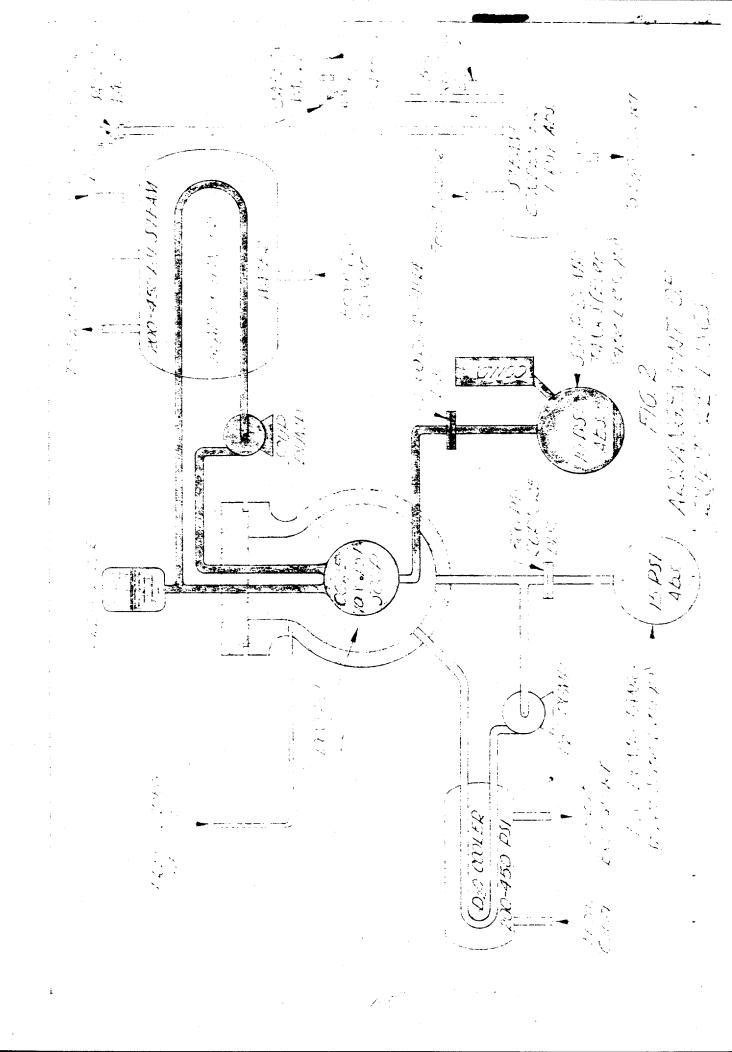
The reflector is automatically dumped 30 seconds after dropping of the safety plates. It will be permissive for the operator to block the dumping at any time after the safety plates are dropped, if in his judgment the situation is sufficiently under control. Under ordinary conditions should the situation not be under complete control, the operator will dump the reflector before the delay time has expired. Dual dump valves are provided for reliability.

The soup dump is a permissive operation by the operator at all times. No automatic dumps are provided. Dual dump valves are provided for reliability.

Stopping of the soup circulation pump is permissive to the operator at all times. The pump will be automatically stopped after the pressure in the steam boiler drops below predetermined levels, say, 20% below the normal operating point. A stoppage of the coolant flow results in a gradual gain in delayed neutrons of about + 0.14% k. The loss in cooling with its accompanying gradual increase in temperature practically compensates for the delayed neutron effect when the natural convection cooling is apparently 1% of the previous reactor cooling. Actual convection losses are calculated to be 20-40 kw. In any event these effects appear to be insignificant. Care should be exercised in restarting circulation, however, once it has stopped. Fortunately calculations indicate that the pump cannot introduce cold (and more reactive liquid) at dangerous rates. However, this is not an operation to be recommended as standard practice.

Upon failure of building electrical supply the safety plates automatically are dropped and soup circulating pump is stopped. Failure of building air supply automatically opens the soup and reflector dump valves. Should cooling water pressure become less than normal an alarm will sound and at a pressure of approximately 20 psi a scram shutdown signal will be given.

2. **Safety Mechanism Association with Process Conditions** (all designed to fail safe). Since the HRE operates at high pressure there are numerous rupture discs set to protect vital parts of the reactor. These discs are shown in Fig. 2 and described below:

A. A 2000 psi rupture disc on a soup system discharging to soup dump tanks.

B. A 1500 psi rupture disc on reflector system discharging to reflector dump tanks.

C. A 600 psi rupture disc on the steam boiler discharging to the turbine condenser.

D. A 50 psi rupture disc on the steam condenser discharging to the atmosphere.

E. 1200 psi relief valves on soup and reflector pump drive system prohibiting the pumps from working against greater heads than 1200 psi.

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FIG 2

ARRANGEMENT OF

In addition there are several additional safety mechanisms as follows:

A. Numerous temperature and pressure switches on process instruments to give scram shutdown signals.

B. A scram shutdown signal initiated if the soup circulation pump stops.

C. Excessive concentration of soup is prevented by condensate overflow line and minimum hold up construction in design of the dump tanks.

Important Interlocks. Soup circulating pump must be operative before the soup feed pump can be started. However, this safety feature may be locked out for initial filling of the core. Soup cannot be concentrated in the dump tanks unless soup feed pump is operating. Soup recirculating pump and soup feed pump cannot operate unless at least one safety plate is withdrawn and held in "cocked" position.

3. **Safety Mechanism and Prevention of the Dissemination of Large Quantities of Activity to the Atmosphere.**

A. Reactor shield ventilation is monitored by ion chambers. Excessive levels of activity shut off ventilating fan (gravity dampers-close), and give scram shutdown signal, and an alarm to the operator to reduce soup system to atmospheric pressure.

B. Building ventilation is monitored by ion chamber providing same set of signals.

III. REACTOR HAZARDS

The HRE is a homogeneous nuclear reactor which operates at high pressure (1000 psi) and moderate power (1000 kw). In principle, therefore, hazardous accidents might result from:

1. Failure somewhere in pressure system without nuclear accident;

2. Hydrogen-Oxygen explosion;

3. Nuclear accident (runaway) resulting in tank rupture;

4. Bombing either by sabotage or attack.

PRESSURE HAZARD

The fact that the HRE operates at 250°C and 1000 psi makes a pressure rupture hazardous since sudden release of pressure will cause part of the soup to vaporize. The pressure tank is really double walled; there is the 3/16 in. stainless steel soup tank which will withstand 4000 psi total, a 3000 psi soup of reflector differential pressure, and the external reflector container constructed to withstand 14,000 psi.

Since failure of any pressure retaining member will cause part of the soup to vaporize, it is imperative that all pressurized vessels and lines be designed and tested with the utmost conservatism. In the following table are listed the working pressure, design pressure, and ultimate bursting pressure for each element of the HRE.

	NORMAL PRESSURE (psi)	PROBABLE MAXIMUM PRESSURE (psi)	ASME CODE DESIGN PRESSURE (psi)	ESTIMATED ULTIMATE BURSTING PRESSURE (psi)
Soup tank	20	1000	-	3000 (differential)
Reflector tank	1000	1020	2000	14,000
Heat exchanger tubes	800	1000	2000	21,800
Lines (soup)	1000	1000	2000	15,500
Soup Dump Tanks	-5	<300	900	5,500
Reflector Dump Tanks	0	385	900	5,500

For the most part, the high pressure soup system is itself enclosed in thick pressure vessels, and the reactor core is surrounded by a very strong reflector pressure vessel, affording a double means of safety.

Two types of failure can be envisaged: first, failure in which the soup system breaks, but the outer pressure system holds; and second, more serious failure in which both systems break. The first failure will ruin equipment; the second will lead to release of fission products into the air.

As far as radioactive hazards are concerned, only the second failure is of importance. To obviate any conceivable chance of a pressure failure releasing radioactivity into the atmosphere, the following measures have been taken;

1. All vessels in contact with the atmosphere have been designed with factor of safety of about 10.

2. The reactor may be completely shut down by *any* of the following procedures, which are non-interdependent:

   (a) Dump the reflector (dump valves in duplicate),

   (b) Dump the soup (dump valves in duplicate);

   (c) Dilute the soup.

   Items (a) and (b) are utilized to reduce the system safely to atmospheric pressure and store the fissionable material in a condition such that further nuclear reaction is impossible under any chain of circumstances. Chances for simultaneous failure of all these means are believed to be extremely remote.

3. Incorporation of rupture discs in reflector and soup systems set for design pressure or less. Rupture discs are well known for their absolute reliability and tamper proof characteristics, and form an almost perfect last ditch automatic safety device. Periodic standard pressure inspection procedures ordinarily applied to fired and unfired pressure vessels will be followed at conservative intervals (probably at least semiannually). This requirement would mean draining the system and application of a hydrostatic test of the reflector and soup system to 3000 psi (with rupture discs removed).

4. (a) Joints in lines containing soup and D_2O will be doubly gasketed with means for detecting leakage through the first gasket, utilizing the second gasket and leak detector tube to confine the leakage to harmless channels (the dump tanks). In the event of a leak in the system in other than a protected joint, assuming the leak is of such magnitude that it is too fast to to salvage much by pressure lowering, but slower than a shield rupturing explosion, one may calculate that 710 cubic feet of

p

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vapor will be formed by the flashing of the hot liquid. The removal of 25,000 Btu would be required for complete condensation. The heat capacity of the equipment and concrete blocks in the shield are certainly far in excess of this requirement. The non-condensable portion formed by the xenon, krypton, etc., will be the order of a few cc. With a sufficiently tight shield, one would plan to dispose of the radioactive gases through the ventilation system provided when favorable weather conditions exist.

(b) The reactor biological shield is constructed and calked as tight to in-leakage (and out-leakage) as possible. A small suction fan will be provided for normal operation, but upon a large increase in stack activity, as indicated by a monitor, the fan will be automatically turned off and the damper closed. Barring a violent explosion, which will rupture the shield itself, the activity from a leak should be retained within the shield, probably being absorbed in the concrete block forming the inner shield.

(c) No effort has been made to design a completely explosive or gas proof building which could be used for the reactor building proper. Standard tight, interlocking construction characteristic of Q type siding will be utilized. The building ventilating system will be monitored, and a heavy overload will similarly shut off the building ventilator and allow the gravity dampers to close. Unless a reactor accident is accompanied by an explosion violent enough to rupture the shield and the building, it is felt that the activity for such occurrence will be spread over a significant period in which natural dispersion, dilution, and decay will give large relative increases in safety over over those situations where the release is relatively uncontrolled.

(d) The reactor foundations form a monolithic saucer or basin capable of holding the entire liquid contents of the reactor. This saucer drains to the active waste storage tank and waste evaporator.

$H_2 \cdot O_2$ EXPLOSION

Radiation damage on the water moderator of the HRE gives rise to gaseous hydrogen and oxygen. If the maximum rate of damage is experienced (50 ev per molecule decomposed) the average life of the water in the HRE at 1000 kw will be $\sim$ 3.3 hours. Since the hydrogen and oxygen will necessarily contain radioactive fission gases, it is felt desirable to recombine the hydrogen and oxygen within the reactor shield, returning the water and fission gases to the system. At the present time, the HRE design has not progressed to the point where firm choices of methods and designs for recombiners have been adopted. However, the probable selection of several known methods is outlined below.

There are four places in the reactor where decomposition can occur, or hydrogen and oxygen can collect, and accordingly recombiners are specified:

(a) Gas from the reactor core.
(b) Gas from the soup dump tanks.
(c) Gas from above the reflector.
(d) Gas from the reflector dump tanks.

It is probable that a catalytic recombiner will be specified for D_2O applications (c) and (d) above. Catalytic recombiners have been operated very safely and successfully by the Canadians at Chalk River, and difficulties are not expected in the application of their experience to our problem.

The gases removed from the core will probably be recombined by thermal combustion in a manner quite similar to that used in the Du Pont heavy water plant at Morgantown. Where large quantities of gases can be expected, this appears to us to be the most favorable and fool-proof method available.

A small amount of gas may form in the vapor space above the soup dump tanks. This gas will eventually collect in the upper end of the soup dump condenser. In view of the possibilities of poisoning a catalytic type recombiner with the fission gases which will also collect in this location, the design will probably specify a thermal or hot wire type of recombiner.

It is planned at this time to provide means for sampling and analyzing (probably by thermal conductivity methods) the gases that collect in each of these spaces. In this way, one will have an independent method for cross-checking the continued efficiency of operation of each of the recombiners.

A violent hydrogen-oxygen explosion in any of these places would probably result in the release of some fission gages to the atmosphere through damage to other equipment within the shield. Specific steps to minimize this hazard beyond the general practice of operating well below the explosive limits for cases b, c, and d, are listed below.

In the case of (c) and (d), we will try to make use of the Canadian experience, which indicates that if the water can be maintained at a sufficiently high purity (conductivity of the order of 1 million ohms) the decomposition will probably not be experienced, eliminating the hazard at its source. An ion exchange column has been planned for the D_2O system in the hope that we will be able to match the Canadian experience, which indicates that recombination is unnecessary.

For case (a), combustion of the flame will be maintained by dual hot wire igniters. and in all probability, a hot ceramic combustion chamber. Suitable temperature indicating devices will be provided for the operator. This recombiner will operate at atmospheric pressure for the following reasons.

(1) To minimize dispersal of radioactivity in the event of a leak.

(2) Although not yet investigated in detail, one stands a good chance of being better able to design a recombiner capable of withstanding an explosion without rupture

(3) With the high pressure drops available in the gas fuel line it is easy to generate sonic velocities (a velocity considerably above flame propagation velocity) through the pressure reading valves in order to insure that a strike-back of the flame cannot get back beyond the reducing valve into the reactor itself.

NUCLEAR ACCIDENTS

The sequence of nuclear, thermal, and hydrodynamic events which follow sudden increases in k of various amounts is treated in great detail in the *Homogeneous Reactor Experiment Feasibility Report*, ORNL 730, and the reader is referred to this report for a complete analysis. We confine our remarks here simply to a brief summary of the situation. Qualitatively, it is clear that after a sudden k jump the reactor power will mount rapidly, this in turn causes a temperature increase which expands the liquid, causing a consequent pressure surge and an ultimate compensating k decrease. The maximum pressure, power, and temperature attained are determined by the size of the k jumps. From the point of view of safety, the essential question is how much k put in at how fast a rate will cause a pressure surge which can be withstood by the system.

As shown in ORNL 630,[12] the following formulae are good approximations to the maximum power, the core soup temperature rise, and the core pressure increase.

$$P_{maximum} = \frac{\lambda_0 (\delta k)^2}{2 S \alpha} \ (kw)$$

$$\Delta Temperature = \frac{2 \delta k}{\alpha} \ (°C)$$

$$Pressure \ increase = (2)(0.0227)\left[\frac{P_{max}}{P_0} - 1\right]^2 \ (psi)$$

λ_0 = Time const. for prompt neutrons $10^4 \ sec^{-1}$

S = Temp. rise per sec per k.w. excess power = .00571

α = Temp. coefficient of reactivity = .001 $°C^{-1}$

P_0 = Equilibrium power

These formulae were derived on the assumption that:

(a) No delayed neutrons are present;

(b) No gas evolution

(c) The system is in a quasi steady state (that is, velocity of fluid is small compared to velocity of sound).

The following table summarizes the values of maximum power, temperature rise, and pressure increase following sudden k jumps at operating temperature of 230°C.

At Temperature of 230°C

$\delta k\%$	$\dfrac{\delta k}{k}\%$	P_{max} (kw)	ΔT (°C)	Δ psi
0.50	0.325	21,200	9.5	21
1.00	0.652	85,000	19	320
1.75	1.135	259,000	33	3020
2.00	1.30	340,000	38	5220
3.00	1.95	761,000	57	26,000

The pressure and temperature rises will be larger as the Δk jump occurs when the reactor is at a lower temperature because $\Delta k/\Delta T$ (temperature coefficient) is smaller. However, the normal operating temperature is approximately 230°C. Tables of maximum pressure, temperature rise, and pressure increase, at temperatures of 150°C and 100°C are given below.

At Temperature of 150°C

$\delta k\%$	$\dfrac{\delta k}{k}\%$	P_{max} (kw)	ΔT (°C)	Δ psi
0.50	0.342	29,700	12	40
1.00	0.685	119,000	24	630
1.47	1.01	258,000	35	3000
2.00	1.37	475,000	48	10,200
3.00	2.05	1,070,000	72	51,700

At Temperature of 100°C

$\delta k\%$	$\dfrac{\delta k}{k}\%$	P_{max} (kw)	ΔT (°C)	Δ psi
0.50	0.35	41,500	16	75
1.00	0.70	166,000	32	1240
1.24	0.87	258,000	40	3000
2.00	1.41	662,000	65	19,800

However, it should be pointed out that:

1. Delayed neutrons will dampen the surge;

2. Gas evolution during the surge will dampen it;

3. No other reactor can stand as large k surge;

4. It is impossible to get sudden k surge;

A consideration of the above leads us to the conclusion that a catastrophic nuclear accident is quite improbable if not impossible.

HAZARD FROM PLANNED SABOTAGE OR FROM A BOMBING ATTACK

Like other reactors, the homogeneous reactor is subject to acts of sabotage which could possibly lead to a hazardous situation. An external explosion adjacent to the reactor could disrupt the reactor core, steel reflector shell, biological shield, and building so that a part of the radioactivity of the reactor would be dispersed. However, as the table on page 35, shows, even if all the reactor contents were thrown into the air, it is very unlikely that any serious radiation hazard would develop either at X-10 or K-25.

Sabotage of the reactor by malicious mishandling of the controls could only lead to an equipment damaging accident and not to a hazard. This type of sabotage is discussed under "non-disastrous accidents" below.

A situation closely analogous to the hazard from planned sabotage exists in the possibility of a bombing attack. Again, the questions of strategic importance of the experimental reactor and its isolated location in regard to possible hazards to other installations and thickly settled areas must be taken into account. The reactor appears to be a poor target from that point of view. A bombing attack would most certainly be under wartime conditions and appropriate measures could be taken to cope with the situation should it become hazardous.

NON-DISASTROUS ACCIDENTS

By premeditated, malicious use of the reactor, its related equipment, and safety mechanism, a well-organized group of saboteurs with enriched uranium available to them could produce an equipment damaging accident. The situation pre-supposes that all the safety devices are made inoperative, that various alarms are ignored, and that the sequence of events is perfectly timed.

Then, should a well organized group of saboteurs be granted sufficient time, manpower, and enriched uranium 235 (~1.5 kilograms) it might be able to break into the enclosed soup system and add uranium to the ~2.0 kilograms already present in the system. This will allow a saturated solution of uranyl sulphate (~770 gms U^{235}/liter) to be attained in the concentrator. Now if the saboteur has blocked or disconnected the multiplicity of safety mechanisms, he will be able to introduce uranium into a cold subcritical core at the rate of ~49 gms/sec by pumping in the saturated uranyl sulphate solution with the Pulsafeeder pump. The photoneutron residual background will probably be around 10^{-6} of normal power level. The power level which will be reached in this situation is ~408,000 kw on the first surge or before the temperature coefficient stops the reaction. The 408,000 kw of power corresponds to ~7550 psi pressure increase in the core vessel. This pressure will rupture the inner vessel as it will take a pressure increase of ~3000 psi before rupture. Upon rupture of the reactor core vessel, the energy would be dissipated in the reflector liquid and the compressible He gas over the liquid. The soup and heavy water reflector upon a pressure rupture would be violently mixed. If all of the uranium from the high pressure soup is included in the mixture the k_c required for criticality is 1.69 at 100°C and 1.65 at 20°C. The material k is 1.56, indicating that upon complete mixing the reactor would be subcritical by a large margin. As the reflector shell is somewhat stronger than the core vessel and the mass of reflector liquid is several times (~8) the mass of soup, an accident or sabotage by this method would very probably never rupture the reflector shell (whose bursting strength is around 14,000 psi).

Note that the supposed saboteur must have a good supply of enriched uranium and must break into the enclosed soup system to develop this accident which is far from being a catastrophe, although some of the equipment is damaged.

These calculations were based only on the negative temperature coefficient
of reactivity, which is large near the operating temperature but somewhat
smaller at room temperature. The negative power coefficient which would be
present if gas bubbles were formed in the soup during power excursion was not
taken into account. The method of calculation the accident is given in
Appendix II.

FLOODS, FIRE, AND WINDSTORMS

From the particular topography selected for the HRE site (see Fig. 1), a
flood, or flood damage, is impossible.

The HRE building as proposed. carries a Uniform Building Code* fire rating
of two hours. Inflammable materials in any quantity are not used in the con-
struction of the building or the reactor, or are any contemplated, except that
the building steam boiler is to be oil fired. The boiler room carries a Uniform
Building Code fire rating of four hours. A 6000 gallon fuel oil storage tank
is to be located 50 feet outside the building. The storage tank is diked in
accordance with the Fire Underwriters Code, and is located such that a leak in
the tank or dike will not drain towards the HRE building. Fire hose cabinets
are provided such that any location in the building is not over 50 feet from
a hose. Fire hydrants on a 6 in. water main are provided outside the building.

The building is designed to withstand wind loads of 20 pounds per square
foot (75 mph) without exceeding the allowable normal working stress of 20,000
psi in the steel structure, in accordance with the New York State Building
Code.

With the enclosure of all og the activity within the sturdy concrete
structure it is highly unlikely or impossible for any reasonable fire, floods,
or windstorm to cause a large release of radioactivity to the atmosphere or
watershed. It is believed that similar considerations apply to the mis-use of
the 7½ ton capacity building crane.

* Pacific Coast Building Operations Conference (AEC Construction Standards per GM-127).

TROUBLESOME SITUATIONS

There are a few features of the homogeneous reactor which may lead to operational troubles or develop into troublesome situations. Under no conditions are these features considered to be hazardous or capable of developing into a hazardous situation.

The production of gas bubbles during operation may lead to some fluctuations as was experienced in the operation of the Los Alamos water boiler. Except for the higher power level and application of pressure to the liquid fuel, the Homogeneous Reactor Experiment is quite similar to the Los Alamos water boiler.

One simple way to estimate the minimum power level at which the HRE can operate without bubble trouble is to compare it with the Los Alamos water boiler. If the gas evolution rate is considered to be the main uncertainty in determining the maximum power attainable in the HRE, then the following comparison is relevant. We assume the HRE will operate as smoothly as the Los Alamos water boiler if the average gas produced per unit volume per second is the same. The water boiler operates at 5 kw, 1 atmosphere pressure, 358°K and has 14 l of solution; the HRE operates at 68 atmospheres, 523°K, and has 50 l of solution. For specific gas volume production to be the same, the HRE must operate at

$$5 \text{ kw} \times \frac{68}{1} \times \frac{358}{523} \times \frac{50}{14} = 830 \text{ kw}.$$

The residence time of the bubbles is assumed the same in the HRE and the water boiler. Actually, since three-fourths of the gas produced in the water boiler is N_2 (which is not present in the HRE), it can be concluded that *if gas evolution limits the maximum power output of the HRE, the Los Alamos water boiler experience indicates that it should be possible to run at a power of $830 \times 4 = 3320$ kw.*

A completely irresponsible operator or a well informed saboteur (without additional enriched uranium available) could by mis-operation of the reactor cause a reactivity excursion but not an accident by introducing over-concentrated uranyl sulphate solution into the reactor core. The worst situation which could be developed would be well below core rupture pressure because of the

limited rate of k introduction and would not lead to a hazard. As in the non-disastrous accident, the situation presupposes that all safety devices fail or are made inoperative, that various alarms are ignored, and that the sequence of events is timed right by accident or planning. The events which could lead to the situation are, reactor is at a subcritical concentration level with soup in the core and reflector in the reflector vessel both at room temperature then, the concentrator concentrates soup in the dump tank system to an optimum to introduce the uranyl sulphate, at a maximum rate the soup pulsafeeder pump is started and introduces concentrated soup into the core. The worst case with the limited amount of uranium in the system will result in a power surge of approximately 146,000 kw which gives a corresponding pressure surge of approximately 960 psi increase over the existing soup pressure in the core vessel. This will not damage the vessel as it will take a pressure increase of approximately 3000 psi before rupture. However, the 960 psi pressure surge may rupture the soup rupture disc in which case the soup flows to the dump tank through the rupture disc drain. None of the soup is lost or dispersed. This problem is thoroughly investigated and discussed in Appendix II.

HEAT AFTER SHUTDOWN

The heat given off by the fission products after the nuclear reaction is stopped has been calculated. The following table summarizes the heat rate and the integrated heat at different times after shutdown:

Time after shutdown	10 seconds	1 minute	1 hour	8 hours	1 day	7 days	30 days
Rate of heat liberation	40 kw	28 kw	12.4 kw	8.0 kw	6.6 kw	4.4 kw	3.3 kw
Integrated heat after 10 seconds	0	0.5 kwh	16 kwh	81 kwh	193 kwh	915 kwh	2940 kwh

The heat values are based on a 1000 kw level of reactor operation for an infinite time and consequently are pessimistic.

A cooling system which will remove heat at the rate of approximately 15 kw (14.2 Btu/sec) will be sufficient, as the mass of the soup system will dissipate some of the initial heat. A cooling water reservoir tank of the order of 100 gallons capacity is constructed inside a corner of the biological shield. Cooling is provided the soup by soldering cooling tubes to the outside of the soup dump tanks. These tubes are connected so as to supply natural convection circulation through the water tank, returning through a bank of fin tubes fastened to the outside of the shield. The bank of tubes will provide sufficient surface area to maintain a cooling rate to the building air of 14 Btu/second. A normally open type valve in the water line prevents circulation during operation of the reactor. Upon failure of the building water supply this valve opens or fails in the open position. A manual by-pass is provided in the event the automatic feature fails.

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EXTENT OF DAMAGE AND RELEASE OF RADIOACTIVITY TO BE
EXPECTED IF FAILURE SHOULD OCCUR

Assuming that a catastrophe has occurred in such a manner that the entire radioactivity of the reactor is disrupted, two cases exist which might cause concern for the safety of individuals in the general locality of the reactor site. This does not include the hazard to the operating personnel who will be endangered to various degrees, depending on the violence of the catastrophe and their location at the time.

There is danger should the catastrophe be such that the radioactivity of the reactor is vaporized and forms a cloud which would irradiate individuals as the radioactive material passes overhead. Some calculations have been made on the amount of accumulated radiation an individual would receive at the X-10 site and outside of the AEC controlled area. This data shows that only under extremely unfavorable weather conditions would X-10 personnel be endangered and then, only after a long warning period had passed. The non-AEC controlled area, being some four times as far from the reactor site as X-10, would very probably never be endangered. These values are tabulated in Table I. Appendix III gives the assumptions on which the calculations were made to arrive at the values.

Another possible danger exists to individuals who depend for their water supply on streams and rivers that are downstream from the reactor site. Again, the danger would arise only if a catastrophe were such that a large part of the radioactivity of the reactor were released and deposited on the ground so that a wash-off of part of the material into downstream water supplies occurred.

By designing the reactor foundation as a monolithic saucer or basin capable of holding the entire liquid contents of the reactor, a considerable portion of the dispersed radioactive liquid can be retained within the reactor building should a destructive catastrophe occur. Of the radioactivity which escapes from the building, part will be airborne and dispersed while part will be distributed over the landscape by splattering and droplet and large particle fall out from the airborne material. This contamination of the landscape will be subject to partial absorption by the soil and partial rain solution leaching. Any hazard to the downriver population will be from that part of the radioactivity which is leached and washed into the river watershed.

TABLE I

Catastrophe: Hazard from Radiation Cloud

(Assuming 100% of activity from 1000 kw operation vaporized)

	Wind Velocity mi/hr	0.1	0.2	0.48	1.0	2.0	3.0	4.0
At X - 10 (0.5 miles)	Accumulated roentgen	179	102	50.8	28.3	16.2	11.7	9.3
	Time from reactor site	5 hrs	2.5 hrs	1 hr	½ hr	¼ hr	1/6 hr	1/8 hr
At 2 miles from site	Accumulated rotentgen	34.0	19.6	9.7	5.4	3.1	2.2	1.8
	Time from reactor site	20 hr	10 hr	4 hr	2 hr	1 hr	2/3 hr	½ hr

There will be ½ hour warning time for 27.4 r dosage and 1 hour for 50 r dosage at X-10. For non-AEC controlled land warning time will be 4 hrs for 10 r dosage.

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To evaluate this potential hazard, safe but realistic estimates of the quantity of activity entering the downstream watershed must be made. To develop the estimates several assumptions were made. These estimates and assumptions are explained more completely in Appendix IV, but briefly they are:

1. A catastrophe which disrupts reactor vessel, reflector shell, and biological shield releasing radioactivity of 1000 kw reactor;

2. Sixty percent of the beta radioactivity finds its way into the downstream water supply,

3. Normal flow of the Clinch and Tennessee Rivers;

4. A silt factor of 0.81 for mixed fission products and 1.0 for Sr^{89} and Sr^{90} (which means that 0.81 of fission product activity is transferred to filtered water)

5. A one day period from time of catastrophe to contamination of Clinch River water;

6. The activity is released in one day;

7. Ingestion is by a water consumption of two liters per day.

The estimated contamination of drinking water and subsequent ingestion of fission products, strontium 89, and strontium 90 are made for the Clinch River immediately below the site and for Chattanooga on the Tennessee River some 124 river miles below the site. These are given in Table II. As the reactor will contain substantially no plutonium, the main danger will be from strontium 89 and strontium 90. Dr. K. Z. Morgan[2] has given tolerance (maximum permissible) amounts for ingestion of 508 microcuries in one day, 2000 microcuries in one year for strontium 89; and 360 microcuries in one year for strontium 90. Drinking water tolerance is 3.3×10^{-3} microcuries per cubic centimeter for one year for strontium 89.

TABLE II

Radioactive Contamination of Downstream Water Supply

Following a Destructive Catastrophe

(Assuming 100% of the accumulated activity added to the watershed 60% of the β activity finds its way into the Clinch River)

	TOTAL FISSION PRODUCT		STRONTIUM 89 40600 CURIES @ T = 0		STRONTIUM 90 667 CURIES @ T = 0		TIME AFTER CATASTROPHE
	FILTERED WATER CONCENTRATION ($\mu c/cc$)	TOTAL INGESTED (μc)	WATER CONCENTRATION ($\mu c/cc$)	TOTAL INGESTED (μc)	WATER CONCENTRATION ($\mu c/cc$)	TOTAL INGESTED (μc)	
Clinch River Contamination	1.86×10^{-2}	37	2.4×10^{-3}	4.8	4.0×10^{-5}	0.08	1 day
Water at Chattanooga contamination	1.5×10^{-3}	3.0	2.6×10^{-4}	0.52	4.5×10^{-6}	0.009	3½ days
Tolerance amount of Sr^{89} and Sr^{90}			By ingestion: 508 μc in 1 day 2000 μc in 1 year		By ingestion: 360 μc in 1 year		

μc = microcuries

cc = cubic centimeters

IV. HAZARD TO SURROUNDING AREA IN EVENT OF A CATASTROPHE

CLIMATOLOGY, HYDROLOGICAL AND METEOROLOGICAL DATA

A detailed meterological report, as kindly prepared by Mr. J. D. Holland of the U.S. Weather Bureau, assigned to the Oak Ridge Operations Office, is attached as Appendix VI.

This report generally concludes that the winds in Bethel Valley blow "up" the valley (winds from west and southwest) 41% of the time; "down" the valley 34% of the time, and only 23% of the time in other directions. The frequencies do not vary greatly between inversion and lapse conditions. There is little reason to believe that the conditions for the site valley differ significantly from those in Bethel Valley.

As given in Section V, the only significant number of personnel within a 2-mile radius are those at X-10. The wind blows toward X-10, 4.4% of the time overall; 3.6% of the time during inversions; 5.2% during lapse conditions. When it is raining, the wind blows in this direction 9.2% of the time. Down in the valley at the reactor site and by analogy with other sites, it is predicted that the wind velocities will be rather low, especially under inversions conditions. If a discharge were to occur under low velocity conditions, the activity would float either directly south from the site or southwest from the site down and follow the descending contours of the land, eventually floating down the already contaminated White Oak Creek valley, and finding an outlet into the Clinch River (Watts Bar) basin. From there it is predicted that the cloud will in general follow the Clinch River through its wanderings until the weather conditions change sufficiently so that lapse conditions exist or the characteristic directional winds appear, at which time greater dilution and dispersion factors apply.

METHODS PROPOSED FOR CONTROL OF STACK GASES AND THE POSSIBILITIES
OF THE CORRELATION OF REACTOR OPERATIONS WITH METEROLOGICAL CONDITIONS

This reactor does not comtemplate the cooling of any of its components with circulated air unless calculations (not completed) indicate that additional cooling for the concrete shield is required. It is planned to make the outer shield as air tight as reasonably possible, and it is believed that the in-leakage with a 1-inch of water suction may be below 1000 cfm and possibly as low as 100 cfm plus the required shield cooling air, if any. There will be some small air spaces within the shield, and accordingly some radio-active argon will be generated. The amount is quite difficult to calculate accurately because of the uncertainty of the air circulation paths, the volume of the void spaces, and the flux in each. Our estimate is that the quantity will be lower by a factor of about 250 than the present X-10 reactor, if a thermal neutron shield is attached to the iron shell. It is planned to discharge this air through a metal stack discharging about 60 feet above the top of the buildings (about 100 feet above the ground level). Should a dangerous burst of activity occur, an air monitor on this line will automatically turn off the exhaust blower and allow a gravity value in the line to close in an effort to prevent gross uncontrolled atmospheric contamination and to confine the bulk of the contamination to the reactor shield itself in the event of a leak in the soup system.

For the small in-leakage rate of 100 cfm the resultant concentration of gases effluent from the stack at 100 feet above the ground will be $\sim 2.0 \times 10^{-4}$ microcuries per cubic centimeter. This is due to the dilution factor only and does not include activity loss from radioactive decay of the argon. The argon does not form any radioactive particulates.

Being an experimental installation, absolute continuity of operation is of rather low importance; and under severe inversion conditions the reactor may be shut down if operating personnel become subjected to unsafe conditions due to argon. These quantities of radioactive argon will be of no hazard to ORNL or other populates under any combination of conditions. It is also feasible—though not desirable—to shut down the reactor whenever the wind is blowing toward K-25 or X-10.

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Certain gaseous fission products, such as xenon, krypton, and probably iodine and bromine, will be generated and removed from the reactor core with H_2 and O_2. As described previously, these will be run through a recombiner, wherein the remaining mixture of steam and fission products will be mixed with those arising from the soup dump tanks and sent to a backwards-return condenser to cool the gases to room temperature. At this time their volume will be of the order of 1.0 cubic centimeter per hour at standard temperature and pressure for the 1000 kw power level. Substantial decay will occur in the second cubic foot vapor space in the condenser. If and when gases are removed from the condenser they will be absorbed into shielded replaceable traps containing activated charcoal. When exhausted, these traps will be replaced and removed to the ORNL burial ground. Through these means, the discharge of radioactive fission products to the atmosphere is effectively prevented.

DATA ON LOCAL EARTHQUAKES, INTENSITY AND NATURE OF
DAMAGE WHICH COULD OCCUR

Data on earthquake frequency in Tennessee from the Fordham University Physics Department indicate that slight shocks have been felt some 14 times in the past 72 years, a frequency of one in five years, approximately. All were slight surface shocks such as are felt everywhere in the world and do not indicate undue seismic activity.

Joseph Lynch, S. J.,[4] of Fordham University states "that it is highly improbable that a major shock will be felt in the area for several thousand years to come."

	1950
Anderson	59,407
Roane	31,665
Knox	233,007
Loudon	23,182
Morgan	15,727

Anderson	Roane	Knox	Loudon	Morgan
Alteringham 20	Byllis	Amherst 250	Martel	Annadel 25
Andersonville 150	Elverton	Asbury 150	LenoirCity	Rugby
Lake City	Webster	Boyd	Bennettown	Breecstertown
Morris	Harriman	Concord	Bucktown River	Burryville
Bethel	Harriman	Ebenezer	Meadow	Huffman
Hinds Creek	Emory Gap	Lyons View	Greenback	Deer Lodge
Knaff	Cardiff	Rocky Hill	Philadel	Sunbright
Offutt	Kingston	Bluegrass	Loudon	Violet Mtt.
Clinton	Rockwood	Cedar Bluff	Erie	Mahan
Beech Grove	Glen Alice	Bearden		Lancing
Briceville	Union Cross Roads	Beaver		Dobey
Medford	Eaton Cross Roads	Byington		Wartburg
Granite	Cave Creek	Amherst		Petros
Chestnut Ridge	Oliver Springs	Croydon		Stephens
Elm pales Ridge	Christman	Freshig		Coalfield
Garden Branch	De Armond	Coster		Blue Gem
Rosedale		Vestal		Oakdale
Braytown		Welwyn		Parker
Marco		Crenshaw		Catoosa
Sugamore		Neuberts		Trankity
Tork Mountain		Ford		Nemo
Indian Fork		Kimberlin Ht		Camp Austin
Windrock		Solway		
Khotan		Virute		
Batley		Fountain City		
Fraine		Powell		
Dossett		Copper Ridge		
Kirkstall		Heiskell		
Peah		Halls Crossroads		
Edgemoor		Bayless		
		Graveston		
		Corryton		
		Maloneysville		
		Mascot		
		Mawell		
		McMillan		
		Beverley		

Knox cont

Loudon	Morgan
Trentville	Kingsley
Thorngrove	Clear Springs
Tuckahoe	White Village
Blade Oak	Marbledale
Whittle Springs	Ramsey
Sunrise	Mule Hollow
Skaggston	Shooks
Middletown	Klondike
Ooddale	Knoxville
Karlsein	
Back Con	
Meadowbrook	
Lonsdale	
Dedie	
Wright	
Westmoreland Hts	
Mt Olive	

Knox cont (Morgan)

Kingsley
Clear Springs
White Village
Marbledale
Ramsey
Mule Hollow
Shooks
Klondike
Knoxville

V. MAKE-UP OF SURROUNDING AREA

DISTRIBUTION OF POPULATION NEAR THE PROPOSED SITE

AEC or Related Employees.

1. Within 0.5 miles of site, only the reactor operating personnel;

2. At 0.6 miles N.W. of site, ~ 2,000 employees at ORNL (X-10) (on a 40 hr week, or 23% of the time);

3. At 1.3 miles N.E. of site, ~ 500 (?) employees of J. A. Jones (on a 40 hr week, or 23% of the time);

4. At 4.8 miles N.W. of site, ~ 5,000 employees of K-25, (on a 40 hr week, or 23% of the time);

5. At 5.2 miles N.E. of site, ~ 2,000 employees at Y-12, (on a 40 hr week, or 23% of the time);

6. At 8 miles N.N.E. of site, ~ 35,000 population of Oak Ridge.

Non-AEC or Related Employees (Population based on late Rand-McNally Atlas)

1. Within 2 miles - none;

2. Within 8 miles - no towns, villages or cities; a few farmers;

3. The following towns within 20 miles:

9 miles North	Oliver Springs	900
9 miles South	Lenoir City	5,000
10 miles Southeast	Concord	Small
12 miles W.S.W.	Kingston	~ 900
13 miles South	Loudon	3,000
14 miles N.N.W.	Petros	1,100
16 miles West	Harriman	6,000
16 miles Northeast	Clinton	3,000
18 miles Northwest	Oakdale	900
19 miles N.N.E.	Briceville	2,500
20 miles East	Western end of Knoxville	---

Between the above towns and the area are located numerous small and somewhat sub-marginal farms with their residents. Because of the large number of changes since the 1940 census it is difficult to estimate the population density of these rural areas in order to arrive at reasonable total population within a given radius.

LIST OF VITAL INDUSTRIAL OR WAR INSTALLATIONS WITHIN
POSSIBLE HAZARD RADIUS OF THE PROPOSED LOCATION

1. ORNL   0.6 miles Northwest

2. K-25 diffusion plant   4.8 miles Northwest

3. Y-12 Electromagnetic plant   5.2 miles Northeast

4. Aluminum Company of America,   22 miles Southeast
    Alcoa Plant

The only TVA dam within 20 miles is the low head Fort Loudon installation (10 miles south).

APPENDIX I

The induced activity in the cooling spray for the reflector tank has been calculated assuming a water flow of 5 gpm and a residence time of 10 sec in a flux of 10^{10}. The approximate composition of the cooling water (filtered river water) is given in Table I.

TABLE I

Water Solids Content

SOLID	CONCENTRATION (ppm)
SiO_2	5.0
Fe	0.05
Al	0.6
Cu	0.02
Ni	0.01
Cr	0.01
Ca	33.0
Mg	9.0
Na	24.0
SO_4	20.0
Cl	4.0
HCO_3	145.0

The activity induced under these conditions is summarized in Table II.

TABLE II

Induced Activity

ISOTOPE	ACTIVITY MILLICURIES PER DAY			
	ZERO DECAY	TEN MIN DECAY	ONE HR DECAY	ONE DAY DECAY
27 sec Oxygen	248	—	—	—
2.3 min Aluminum	11	0.5	—	—
14.9 hr Sodium	2.9	2.9	2.8	0.95
38 min Chlorine	2.2	1.8	0.7	—
9.5 min Magnesium	0.7	0.4	—	—
5 min Sulfur	0.4	0.1	—	—
TOTAL	265.2	5.7	3.5	0.95

$$\text{Flow} - 5 \text{ gpm} = 2.76 \times 10^7 \text{ cc/day}$$

$$\text{Activity} = \frac{5.7 \times 10^3 \ \mu c/day}{27.6 \times 10^7 \ cc/day} = 2.07 \times 10^{-4} \ \mu c/cc$$

This activity level does not appear to warrant any holdup measures before discharge.

APPENDIX II

INVESTIGATION OF THE POSSIBILITY OF ACCIDENTS DUE TO
RAPID INCREASE OF SOUP CONCENTRATION

A study has been made of the feasibility of producing an undesirable reactivity excursion by manipulation of the reactor controls. Many operating conditions have been considered, and the most serious of these is presented below. This condition does not constitute a reactor hazard in the usual sense, and would result, at most, in an extended delay for repairs. It should be noted that careful planning would be required to produce this condition, to the extent of sabotage or the equivalent.

The most serious excursion may be produced by increasing reactivity as rapidly as possible, through the critical point, when the reactor is clean and cold The smaller the neutron source, prior to criticality, the greater the resulting reactor excursion.

The maximum rate of reactivity change may be accomplished by increasing the soup concentration in the dump tanks to a maximum (under specific conditions) and then pumping it into the sub-critical core at the maximum pumping rate of 1 gpm The effects of changing reflector level, withdrawing absorbing plates, or changing the temperature of the core, or reflector, are relatively small since they are limited by the synchronous speed of electrical motors, or by thermal capacity.

Several qualified scientists of critical assembly experience will be present for the first startup and each succeeding *clean and cold* startup thereafter (if any). It is assumed that the reactor is unavailable for sabotage during this period. It should be pointed out that the quantities and half-lives of the activity resulting from a mistake at this time must also be quite small. It is assumed that any time thereafter (unless the soup is drained and replaced with fresh inactive soup—requiring the saboteur to possess a large source of enriched uranium) that there is sufficient residual gamma activity to provide a source of neutrons of the order of 1 watt by virtue of the $(\gamma - n)$ reaction on the D_2O. This seriously and automatically limits the damage possible by sabotage in gross-misoperation.

The reactor system is assumed to be at 20°C. After any significant period of operation the heat-after-shutdown will probably limit the minimum temperature of the reactor to 60°C with mis-operation, or about 100°C with normal operation. Either of these higher initial temperatures lessen the hazard.

The effect of most serious excursion has been calculated using the following assumed conditions.

(a) All safety devices are inoperative.

(b) The system is charged with 2100 gms of U^{235}. No additional U is available to the operator and the filling valve is locked closed.

(c) The reactor is subcritical with soup circulating through the core and heat exchanger at room temperature. The worst excursion results if this system contains $\sim$ 1050 gms of U in 60 liters (1300 gms are required for criticality under these conditions; $i.e.,$ 1080 gms in the core).

(d) The reflector is full and at room temperature.

(e) The high pressure feed pump (Pulsafeeder pump) is stopped and the soup in the dump tanks is concentrated to $\sim$ four liters by evaporating water and holding it in the condensate tanks.

The feed pump must be stopped if an appreciable concentration difference is to be built up between the tanks and the reactor core. The concentration in the core, heat exchanger, feed, line (1 liter) and filter (1 liter) is: 1050/60 = 17.5 gms U/l. This leaves 2100 gms - 17.5 × 62 liters = 1018 gms U in the dump tanks which has been concentrated to 1018/4 liters = 254.5 gms/l.

(f) At this precise point the Pulsafeeder pump must be started and operated at its maximum speed of 1 gpm = 0.0634 l/sec. The dilute liquid in the pump line and filter will be pumped into the reactor first, and will be followed by a "slug" of concentrated solution. The dilute material must be held in the pressurizer for $\sim$ 16 sec, for if the automatic level control in the pressurizer releases it to the dump tanks, the concentrated material will be diluted.

As the concentrated slug starts into the core, concentration begins to increase at a rate of

(254.5 - 17.5) gms/l × 0.0634 l/sec = $\sim$ 15 gms/sec.

Criticality will be reached when 1300 - 1050 = 250 gms have been added to the system. This requires

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$$\frac{250 \text{ gms}}{15 \text{ gms/sec}} = \sim 16.6 \text{ sec}$$

after the slug started in, or

$$\frac{2 \ l \ (\text{pipe and filter})}{.0634 \ l/\text{sec}} + 16.6 \text{ sec} = \sim 48.6 \text{ sec}$$

after the pump started.

The elevation of the feed pump section pipe is arranged so that the feed pump cannot remove liquid unless more than three liters are present in the dump system. Criticality cannot be reached unless the "slug" of concentrated material is $\sim$ 250 gms. Therefore a slug of 250 gms/(254.5 - 17.5) gms/l = $\sim$ 1.06 liters must be pumped. If the soup is originally concentrated to a volume < (3 + 1.06) = 4.06 liters the reactor will not become critical. If the concentrated volume is > 4.06, the concentration is greatly reduced, resulting in a decreased reactor excursion.

Example: If the volume is 4.06 liters the rate of concentration increase in the core is $\sim$ 15 gms/sec; for 8 liters the rate is 6.95 gms U/sec.

It should be noted that some solution must be drained into the dump tanks from the core between 16 and 32 sec after the pump is started, otherwise the solution is diluted or the Pulsafeeder pump will become steam-bound.

DISCUSSION OF RESULTS

If all of the above precise conditions are met, it is possible to obtain a rate of concentration of $\sim$15 gms U/sec. Assuming an initial neutron source equivalent to one watt* the maximum power excursion will be $\sim$ 118,000 kw, and the corresponding maximum pressure increase will be $\sim$ 620 psi (using the equations which follow). The system is capable of withstanding a pressure increase of $\sim$ 3000 psi and therefore is entirely safe. As mentioned elsewhere in this report, a rupture disc is provided which will rupture at 1000 psi pressure increase, and this would operate if higher pressures were attained.

The calculations are based only on the negative temperature coefficient of reactivity. The negative power coefficient which would be present if gas bubbles were formed in the soup during the power excursion was not taken into account. This gives additional safety and tamper-proof protection to the reactor.

EFFECT OF CONCENTRATION AT HIGHER TEMPERATURE AND POWER

At higher temperatures the effects of increasing reactivity as above are very much lessened, due to the much higher temperature coefficient of the reactor and smaller amount of excess uranium available for concentration in the dump tanks. (Approximately 1700 gms are required in the high pressure system for criticality at 200°C as compared to ~ 1300 gms in the previous case) This leaves only 340 gms of uranium in the dump tanks when just critical at 200°C and the maximum concentration rate is ~ 7.15 gms/sec as compared to the previous rate of 15 gms/sec.

The temperature coefficient increases from $a = 0.00019$ δk/°C @ 20°C to 0.001 δk/°C @ 200°C. If the reactor is operating at 1000 kw when the concentration in the core is changing at the above rate (7.15 gms U/sec) the maximum power excursion is calculated to be ~ 22,600 kw, and the pressure increase is ~ 21 psi.

EQUATIONS AND DATA

The following equations were used for calculating power excursions and pressure increase.

The equation for calculating *power excursions* is:

$$(1) \qquad \frac{C}{aS} + P_H = \frac{P_{max} - P_i}{\ln P_{max} - \ln P_i}$$

where $C = \delta k$/sec; a = temperature coefficient of reactivity; S = temperature rise per sec per kw excess power; P_H = power draw-off in kw; P_{max} = maximum power in kw; P_i = initial power kw δk is found from the equation:

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$$k = \frac{N}{1 + 0.0128 \dfrac{WH_2O}{WU}}$$

where $N = 2.09$; WH_2O and WU = weights of H_2O and U^{235} in 60 liter system. WH_2O is an essential constant at a given temperature and

$$\frac{\delta k}{\delta WU} = \frac{0.0128 \ WH_2O \ (k)^2}{2.09 \ (WU)^2} ,$$

At 20°C soup and reflector $k = 1.31$, and at 200°C soup and 100°C reflector $k = 1.50$. At 20°C, $\alpha = 0.00019 \ \delta k/°C$; $S = 0.0048°C/kw$ sec. At 200°C, $\alpha = 0.001 \ \delta k/°C$; $S = 0.0054°C/kw$ sec.

Initial power, P_i, will be $> 10^{-6}$ of normal operating power for at least 72 hours after operation due to $(\gamma - n)$ reaction of fission products gamma rays with the heavy water reflector to produce a residual background of photo-neutrons.[5,6] An artificial source will be inserted for the first critical experiments.

The equation for *pressure increase* is:

$$(2) \quad psi(max) - psi_0 = (2)(0.0227)(P_{max}/P_0 - 1)^2$$

where $psi_0 = 1000$ psi, $P_0 = 1000$ kw the equation by derivation gives a constant of 0.0227, and by numerical integration produces a constant twice as large. Equations (1) and (2) are developed in ORNL 632, *Equations of Motion of the Homogeneous Reactor*,[7] to be published soon.

APPENDIX III

CALCULATION OF THE AMOUNT OF RADIATION FROM A CLOUD
FOLLOWING A CATASTROPHE

For the estimated radiation dosage an individual would receive in the general location of the reactor site, the following assumptions were made:

1. A disruptive catastrophe which disperses all of the 1000 kw reactor's fission products into the air;

2. The resultant radiation cloud has a thickness of 1500 feet;

3. The following equation[8] is used:

$$\text{Accumulated roentgens} = \frac{2 \times 10^{10} \times \text{Power (kw)}}{[\text{cloud thickness (cm)}][\text{wind velocity (cm)}]^{0.8} \; [\text{distance from reactor (cm)}]^{1.7}}$$

Cloud thickness (1500 ft) $= 4.5 \times 10^4$ centimeters

Distance from reactor $= 7.9 \times 10^4$ centimeters for X-10

$= 3.16 \times 10^5$ centimeters for 2 miles

Wind velocity, 1 mile/hr $= 44$ centimeters/sec

APPENDIX IV

DATA AND INFORMATION TO MAKE AN ESTIMATE OF DOWNSTREAM
WATER CONTAMINATION FOLLOWING A CATASTROPHE[9]

The average river flow rates at several TVA dams—all except Norris are on the Tennessee River—and the distance (in river miles downstream) of each from White Oak Creek (watershed of the site) are as follows:

DAM OR LOCATION	AVERAGE FLOW $(ft^3/sec.)$	DISTANCE
Norris	4,100	-59.2*
Watts Bar	26,400	58.4
Chickamauga	36,500	117.3
Chattanooga	~	124.2
Hales Bar	38,000	164.1
Guntersville	42,000	246.2
Wheeler	49,000	320.3
Wilson	50,500	335.8
Pickwick	54,000	388.5
Kentucky	65,000	572.8
Paducah	~	595.2

* Upstream on Clinch River

The estimate of 60% of the beta radioactivity which finds its way into the water supply is obtained by considering that one third of the 20% possible airborne will fall out in the watershed, and that of the remaining 80%, at least a third will stay behind in the building basin or debris in such a manner that its release can be prevented.

The normal flow of the Clinch and Tennessee Rivers was used to calculate the dilution of activity concentration. Should the flow be below normal the

rate of release of the contamination to the river would be slower. Should the rate of release be faster, the river flow would be larger and consequently greater volume dilution would be attained. The stream velocity was taken as two miles per hour to take the contamination to Chattanooga.

The silt factor is estimated from experience at White Oak Lake. Ordinarily much more than 30% of the solution activity will by natural clay ion exchange absorb on the normal waterborne silt burden. A reasonable guess may consider 40% of silt to settle. Of the unsettled silt, part may be released by water processing plants downstream. A safe value may be 60% of activity released to solubility from the unsettled silt. Thus we have at first: 70% in solution, 30% in silt. Of the latter, 40% settled, leaving 18% of the original total. Sixty percent of this may go back into solution when treated, adding 10.8% to the initial 70%, which gives a factor of 0.81. This 0.81 factor was not used for calculations on strontium.

The beta activity of fission products was calculated by the equation

$$\beta \text{ radiation} = 27 \times 10^{13} \ t^{-0.26} \ P \text{ Mev/sec,}$$

where t is time in seconds, P is reactor power kw. One Mev per disintegration for beta was assumed, and one curie $= 3.7 \times 10^{10}$ disintegration/sec gave conversion to curies.

River flow for Clinch:

$$(4100 \text{ ft}^3/\text{sec}) \ (2.83 \times 10^4 \text{ cc/ft}^3) \ (8.64 \times 10^4 \text{ sec/day}) = 10^{13} \text{ cc/day}$$

For strontium, fission yield and half-lives of 55 days for Sr^{89} and 25 years for Sr^{90} gives:

TIME	CURIES OF Sr^{89}	CURIES OF Sr^{90}
0	40,600	667
1 day	40,000	667
3½ days	38,800	667

APPENDIX V

CALCULATIONS OF STACK GAS ACTIVITY AND VOLUME OF RADIOACTIVE FISSION

PRODUCT GAS TO CHARCOAL ABSORBER

Stack Gas Activity. Radioactive A^{41} is produced in the shield from atmospheric argon. An estimation of the activity of gas in the shield due to the A^{41} production is made by assuming:

1. The total number of neutrons per second flowing into the shield is 0.6×10^{16}.

2. All neutrons are absorbed in the shield.

3. Ten percent of shield is air voids. Air is at atmospheric pressure 60°C and containing 0.8% argon. Absorption cross section of argon is 1.2 barns.

4. Ratio of absorption in argon to total absorption is 0.22×10^{-5}.

The production of A^{41} is 1.3×10^{10} atoms per second, or 3.3 curies per day. At 1000 cfm the concentration at the stack exit would be 0.8×10^{-4} $\mu c/cc$. The production of A^{41} in the X-10 reactor is 3.15×10^{12} or 800 curie per day. The stack exit at the X-10 reactor is 10^{-4} $\mu c/cc$.

Radioactive Fission Product Gases. Report CF 49-9-114, *Production of Gaseous Fission Products in Homogeneous Reactor,*[10] gives volume of production of Br, Kr, I, and Xe as 202 mm³ per hour @ STP for 200 kw level.

The volume for 1000 kw level is 1010 mm³/hr (5 × 202), as the relation with power is direct, 1010 mm³/hr $\cong$ 1 cc/hr.

APPENDIX VI

CLIMATOLOGY DATA—WINDS IN THE VICINITY OF X-10 CORRELATED WITH THERMAL STABILITY AND PRECIPITATION

U. S. Weather Bureau
Oak Ridge, Tennessee
March 21, 1950

Winds at the 140 ft Water Tower in Relation to Stability. Table I shows the average annual frequency of wind direction to eight points at the wind instruments on the 140 ft water tower operated by ORNL Health Physics Division at X-10 for the period January, 1944, through February, 1950. Frequency distributions are given for all hours with zero or positive temperature gradient from the 4 ft to the 183 ft level on the X-10 water tower ("Inversion"), all hours with negative temperature gradients ("Lapse") and all hours combined. Figures 1, 2 and 3 show the data in wind-rose form.

TABLE I

Wind Direction Frequency (Percent)

	INVERSION	LAPSE	TOTAL
NE	28.2	21.6	25.0
E	9.8	8.1	9.0
SE	3.6	5.2	4.4
S	7.0	5.5	6.3
SW	21.1	30.2	25.4
W	16.2	15.5	15.9
NW	5.1	8.8	6.9
N	6.4	4.2	5.3
Calm	1.6	0.5	1.0
Indeterminate	1.0	0.6	0.8

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The prevailing wind direction is west-southwesterly, with a secondary maximum frequency from the east-northeast. These appear as large percentages of west, southwest, east and northeast in the eight-point rose, and reflect the orientation of the broad valley between the Cumberland Plateau and the Great Smoky Mountains on a large scale, and on a smaller scale, of Bethel Valley (which contains the X-10 site), and the adjoining ridges. The WSW wind is the up-slope wind, the ENE wind down-slope, in relation to the larger valley. Over a period of approximately six years, the X-10 wind direction instrument has recorded W or SW 41% of the time, and E or NE 34%, with only 23% from the remaining 180° of the compass. During inversion (zero or positive vertical temperature gradient in the lowest 180 feet) the E and NE winds show a slight predominance, occurring 38% of the time as compared with 37% from W and SW. On the other hand, in lapse conditions (negative vertical temperature gradient in the lowest 180 feet) W and SW occur 46% of the time as compared with only 30% from E and NE. The west-southwesterlies are also the predominant winds of spring and early summer, and of the afternoon hours, while the east-northeasterlies prevail in late summer and fall, as well as during the night and early morning hours. In winter this local circulation is overshadowed by the frequent west winds associated with moving weather systems. Thus, in general, the westerly and southwesterly winds are the more turbulent winds and the northeasterlies the more stable winds.

The annual average speed of the wind at the 140 ft tower is 5.7 mph, varying seasonally from about 4 mph in summer to about 7 mph in winter. Calms occur only 1% of the time at this height, the frequency being greatest (1.6%) during inversions and least (0.3%) during lapse. Maximum speeds occur during the afternoon hours, with the lowest average speed and greatest frequency of calms occurring at night.

Frequency of Inversions. Inversions as defined in the first paragraph occur about 52% of the time annually. The monthly and annual frequencies for the period of record are shown in Table II. It can be seen that while there are wide variations from year to year, the greatest monthly frequency tends to occur in October with an average of 61% and the least in June averaging 48%.

This monthly distribution of stability and instability in the lowest 180 feet of the atmosphere is correlated with conditions in the deeper layers

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TABLE II

Frequency of Inversions (Percent)

	1944	1945	1946	1947	1948	1949	1950	AVERAGE
January	42.5	42.5	59.0	53 1	64.4	18.4	61 5	48.8
February	25.1	30.2	61.5	M	69 0	50 7	62.1	49 8
March	45.5	44.9	61.5	M	61.0	61.9		55 0
April	40.9	43.2	58.3	M	55.5	52.1		50.0
May	49.9	46.9	46.5	M	48.3	62.6		50.8
June	42.4	47.9	48.3	39.7	49.7	56.9		47.5
July	48.4	50.6	49.3	51.8	41.8	61.2		50.5
August	41.5	59.0	53.7	67.0	31.8	53.1		51.0
September	49.6	62.1	57.4	51.4	33 4	62.6		52.8
October	60.4	60.2	67.1	77.6	30.7	71 2		61.2
November	43.2	74.3	42.5	57.1	36.0	70.0		53.9
December	62.6	56.6	38.8	75.8	31.8	62.2		54.6
Annual	46.0	51.5	53.7	59.2	46.1	56.9		52.2

M - Missing

of the troposphere, as evidenced by the 79 years of standard weather observations at Knoxville, approximately 25 miles east of Oak Ridge. If the months are ranked from 1 to 12, in descending order, according to high frequency of clear skies, dense fog and low visibilities, low frequency of partly cloudy skies (indicative of turbulence) and thunderstorms, high daily temperature range and 1:30 p.m. relative humidity (both indicative of restricted vertical mixing) and low average wind speed, the sum of ranks should be a fair stability index. This type of analysis shows October, November and December to be the most stable (lowest total rank), July, June and March the most unstable months.

Furthermore, a high pressure center on the monthly mean sea level pressure charts, indicative of a tendency toward stability in the lower few thousands

of feet, occurs in this portion of the country during October, November, December and January. Thus, although the total frequency of temperature inversions will become considerably less above 500 to 1000 feet due to the disappearance of the nocturnal radiation inversions, the seasonal variation will follow a similar pattern at the greater heights.

Local Variations in the Winds. Considerable variation is observed in both wind speed and direction within small distances in the valley, as a result of variations in elevation, slope and vegetation cover. Instruments located in relatively sheltered places such as White Oak Creek Pass through Haw Ridge, the adjacent low areas of Bethel Valley, and areas covered with tall trees show, in the lowest 50 feet, 20 to 90% of calms at night (8 p.m. to 8 a.m.) and annual average speed of less than 3 mph. Well exposed valley locations (rises within the valley, free of trees) have 10 to 20% of calms at night, few calms in the daytime, and average speeds in the neighborhood of 4 mph. Exposed hill-tops have very few calms at any time and average speeds of 4 to 8 mph, comparable to those observed on the water tower.

Up-slope wind directions prevail in the lowest 50 feet at points off the valley axis during daytime, lapse, and early summer, accompanied by gustiness and moderate speeds. Light downslope movements prevail at night and in inversions. Bethel Valley and the adjoining ridges form the dominating configuration with respect to the air flow. Such secondary features as the White Oak drainage basin within Bethel Valley, the many ravines in the ridges, and even the White Oak Creek gap in Haw Ridge make a negligible contribution to the total transport of air. Although these features may produce pronounced local eddies, such effects do not extend far towards the center of the valley.

Within this surface layer horizontal continuity of movement is poor, especially in cross-valley directions; each valley may develop an independent circulation, the major outlet being through exchange with the upper air via the slope winds and convective eddies. Only with strong winds or with prevailing upper winds (2000 to 5000 feet) parallel to the valleys would it be of value to extrapolate air movements for any number of miles using valley wind observations. An exception is the well developed inversion case where even a very light air movement will follow the valley as far downstream as the valley retains its structure even though the prevailing winds a few hundred feet above the ground may be in an entirely different direction. In general, air transport

from a valley location will be governed by the valley wind regime combined with the mode and degree of exchange with the upper air, and the winds in the upper streams.

Upper Winds. Table III and Figs. 4 and 5 show the frequency distribution of wind speed and direction to 16 points at 3000 and 10,000 ft above sea level at Knoxville, Tennessee, based on 9162 pilot balloon observations at 3000 ft and 4919 observations at 10,000 ft. These wind roses represent all the observations prior to January 1, 1939, and are derived from the U. S. Weather Bureau Airway Meteorological Atlas for the United States.

TABLE III

Frequency of Wind Speed and Direction (Percent)

	3000 ft Above Sea Level					10,000 ft Above Sea Level			
	Under 18 mph	18-31 mph	Over 31 mph	Total		Under 18 mph	18-31 mph	Over 31 mph	Total
NNE	4	1	0	5		2	1	0	3
NE	4	1	0	5		1	1	0	2
ENE	4	2	0	6		1	1	0	2
E	2	1	0	3		1	0	0	1
ESE	1	0	0	1		1	0	0	1
SE	1	0	0	1		1	0	0	1
SSE	1	0	0	1		2	0	0	2
S	2	1	0	3		2	1	0	3
SSW	4	3	1	8		3	2	0	5
SW	6	8	3	17		4	3	1	8
WSW	7	6	1	14		4	4	3	11
W	6	4	1	11		5	7	6	18
WNW	4	2	0	6		4	7	6	17
NW	4	1	0	5		4	5	3	12
NNW	3	1	0	4		3	3	1	7
N	4	1	0	5		2	2	1	5
Calm				6					2
Total	57	32	6	101		40	38	21	100

Above the ridge tops, the winds follow closely the broad currents of the main valley between the Cumberlands and the Smokies. From the 3000 ft level (2000 ft above ground) there is, on the average, no significant difference between observations taken at Oak Ridge and those taken at Knoxville.

The northeast-southwest axis of the large valley continues to influence strongly the direction distribution up to about 5000 ft, the southwesterly winds increasing steadily in frequency at the expense of the northeasterly. Above 5000 ft the southwesterly winds give way to the prevailing westerlies observed generally at this latitude. At 2000 ft above the ground the 90° sector centered between southwest and west-southwest contains 50% of all observations. By 10,000 ft 58% of the observations show directions within the quadrant centered between west and west-northwest.

The greatest frequency of surface wind direction coinciding with or approximating the upper wind direction occurs in lapse conditions. In inversions the lower layers tend to move in a direction independent of the upper wind.

The average speeds increase from 1-4 mph within 50 ft of the valley floor (800 to 900 ft above sea level) to 6-8 mph at ridge top level (1100 to 1400 ft above sea level), then to about 15 mph at 3000 ft and 25 mph at 10,000 ft. The frequency of winds less than 4 mph correspondingly decreases from over 50% near the valley floor to about 30-40% at the highest anemometer stations, 6% at 3000 ft and 2% at 10,000 ft.

Correlation of Wind Direction with Precipitation. Table IV and Figs. 6 and 7 show the frequency distribution of gradient and 10,000 ft wind direction to eight points for times with rain at Oak Ridge. These wind directions were obtained from the isobars in the published U. S. Weather Bureau Daily Weather Maps for August, 1948, through February, 1950. The 10,000 ft wind directions are from the 700 mb contours at 10 a.m. on days during which rain was recorded. The gradient wind directions are from the sea-level weather maps with rain at the time of observation. These latter directions should compare well with the actual wind directions at 2000 to 4000 feet above sea level.
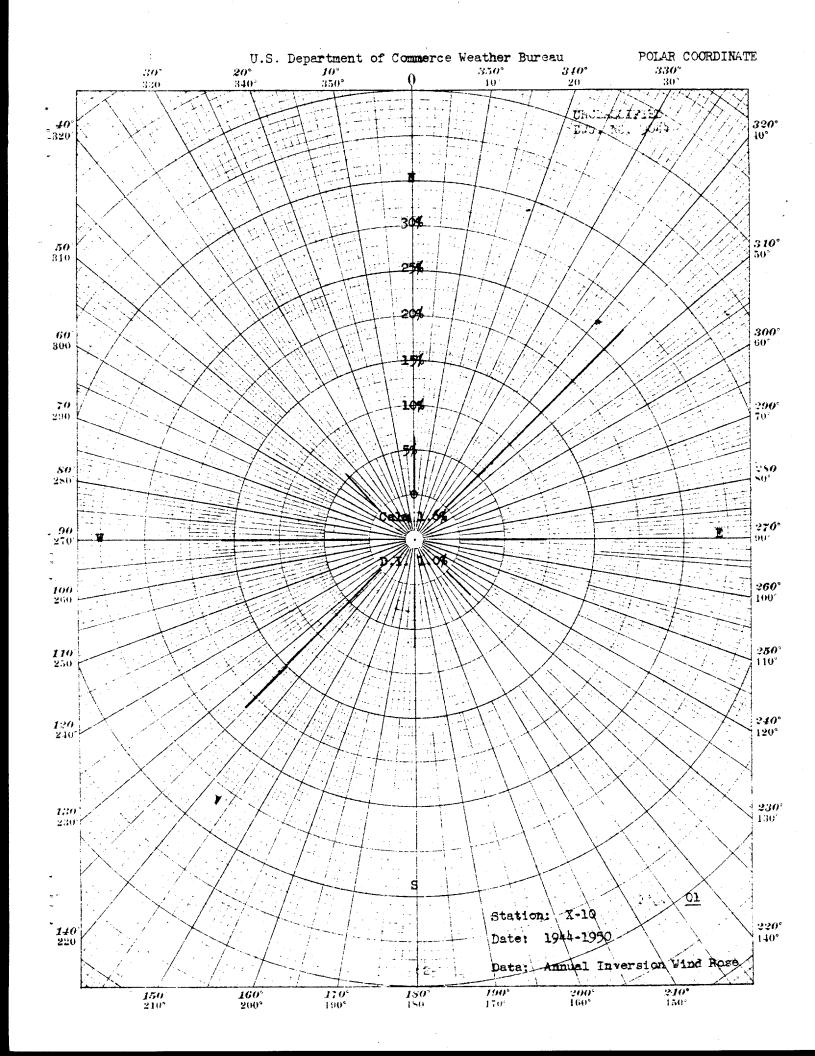
It is seen by comparison of Figs. 6 and 7 with Figs. 4 and 5 that the rain wind rose resembles the average pilot balloon wind rose for the corresponding level shifted approximately 45° to the south. The gradient level now shows 46% of south and southwest combined, and the 10,000 ft level 75% of
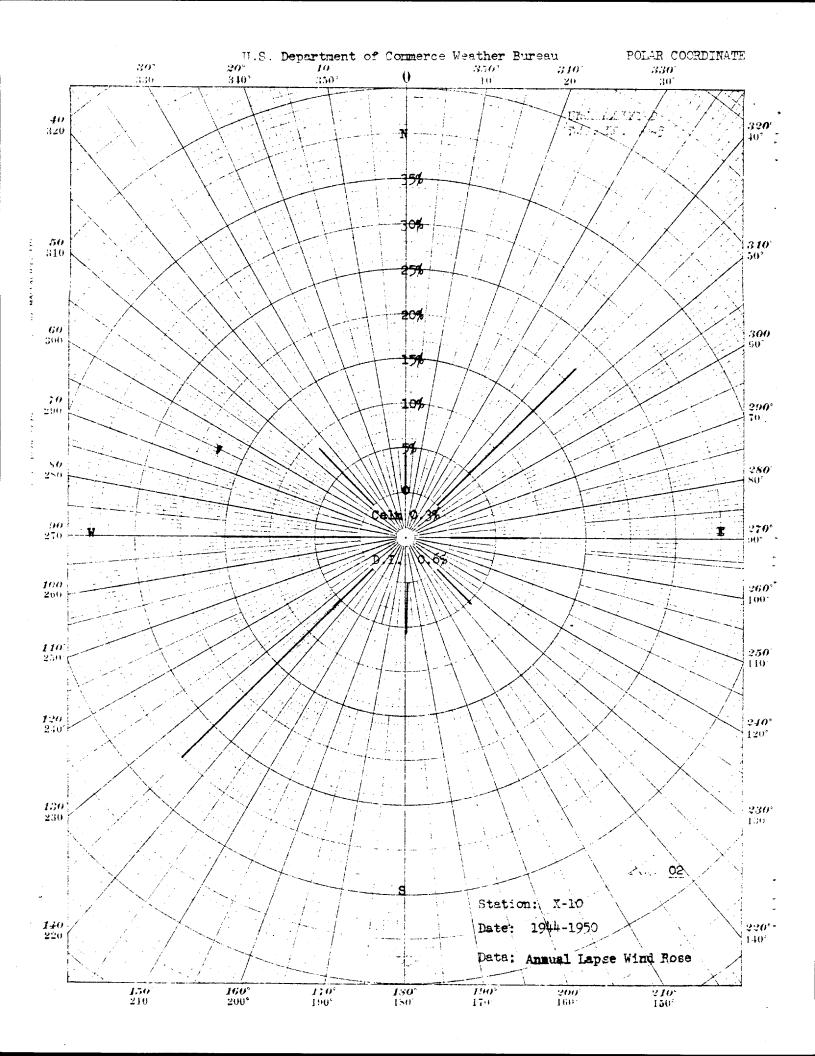
southwest and west.
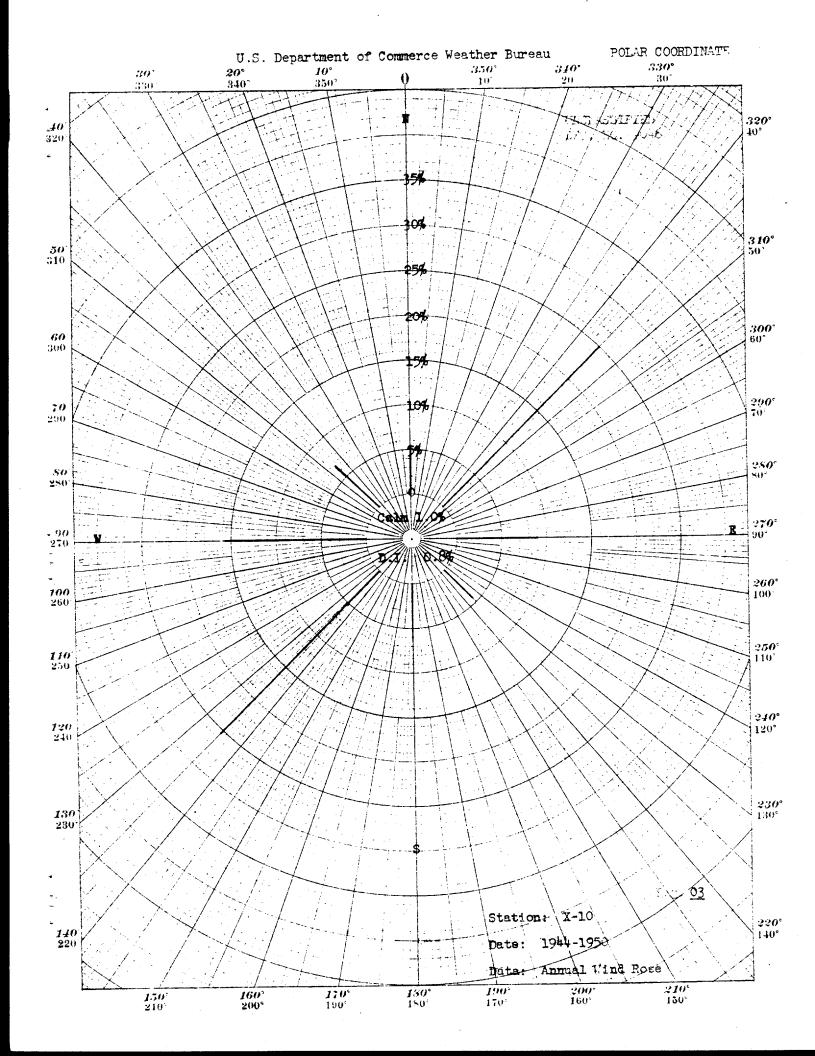
The annual number of days with .01 in. or more of precipitation, based on 79 years of record at Knoxville, is 134. The greatest frequency, 13 days per month, occurs in January and March, the least 7 to 9 in fall. In the short period of record at X-10, it appears that the precipitation frequency may be slightly higher than that at Knoxville, but the difference is not significant by comparison with the variations between different years at the same station.
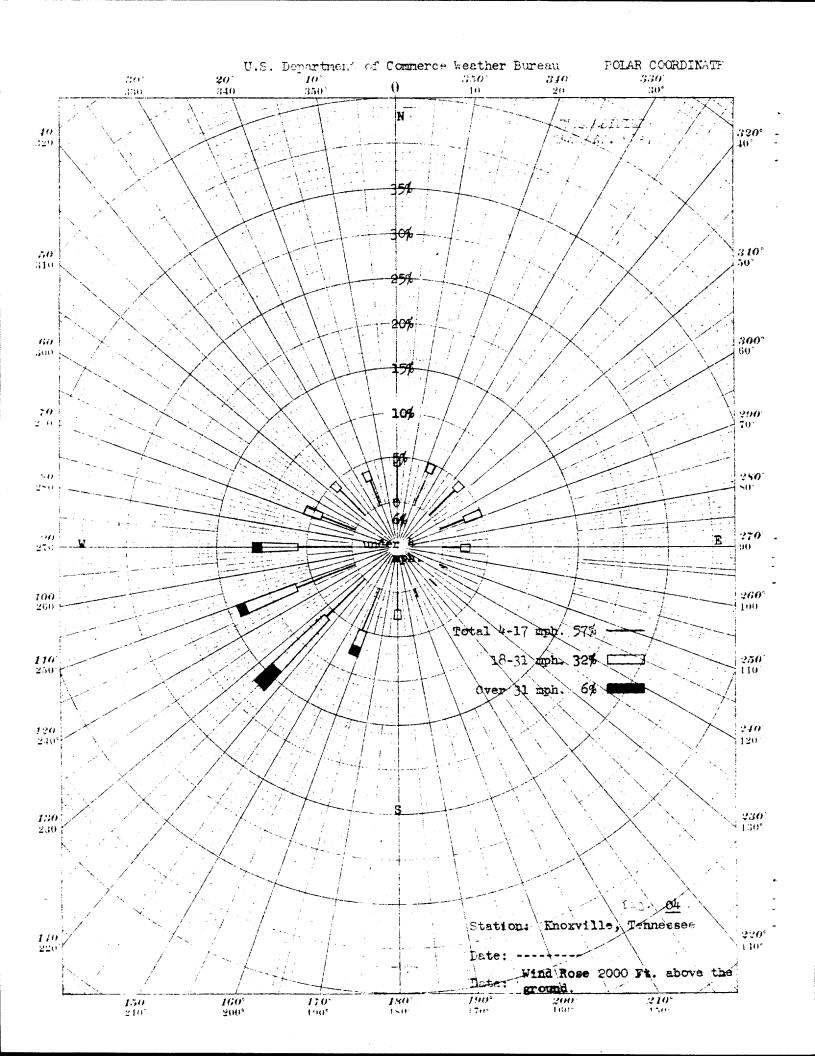
TABLE IV

Wind Direction Frequency with Rain (Percent)
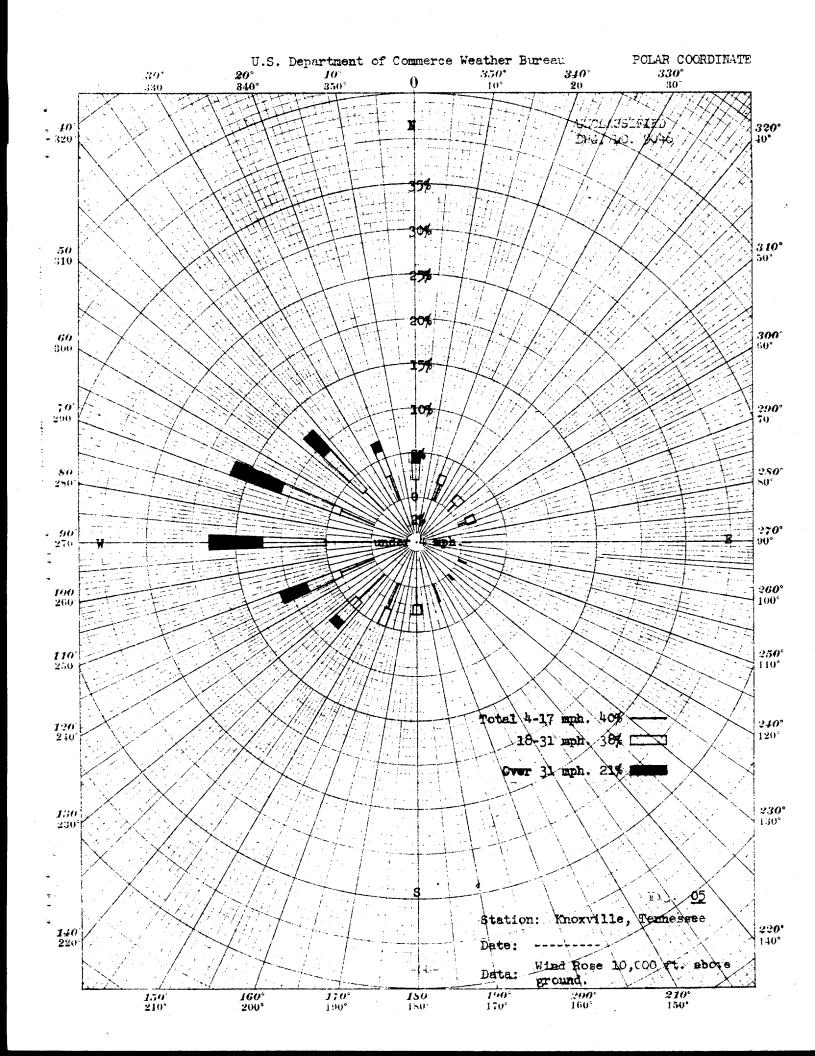
100% =	GRADIENT		10,000 FEET	
	RAIN OBS.	ALL OBS.	RAIN OBS.	ALL OBS.
NE	10.8	3.6	0	0
E	3.6	1.2	1.1	0.5
SE	9.2	3.0	5.0	2.3
S	20.9	6.8	6.5	3.1
SW	25.2	7.9	45.2	21.4
W	8.4	2.7	29.8	14.2
NW	9.3	3.1	10.7	5.1
N	10.7	3.6	1.5	0.7
Calm	1.9	0.6	0	0
Total	100.0	32.5	99.8	47.3

Station: X-10

Date: 1944-1950

Data: Annual Inversion Wind Rose

30°
330°   20°
340°   10°
350°   0   350°
10°   340°
20°   330°
30°

40°
320°

50°
310°

60°
300°

70°
290°

80°
280°

90°
270°

N

35%

30%

25%

20%

15%

10%

5%

Calm 0.3%

W

D.I. 0.65

100°
260°

110°
250°

120°
240°

130°
230°

140°
220°

320°
40°

310°
50°

300°
60°

290°
70°

280°
80°

270°
90°

260°
100°

250°
110°

240°
120°

230°
130°

220°
140°

S

02

Station: X-10

Date: 1944-1950

Data: Annual Lapse Wind Rose

150°
210°   160°
200°   170°
190°   180°
180°   190°
170°   200°
160°   210°
150°

U.S. Department of Commerce Weather Bureau   POLAR COORDINATE

Station: X-10

Date: 1944-1950

Data: Annual Wind Rose

Total 4-17 mph. 57% ———
18-31 mph. 32% ☐
Over 31 mph. 6% ■■■

Station: Knoxville, Tennessee

Date: ----------

Date: ---------- Wind Rose 2000 Ft. above the ground.

Total 4-17 mph. 40%
18-31 mph. 38%
Over 31 mph. 21%

Station: Knoxville, Tennessee

Date: ----------

Data: Wind Rose 10,000 ft. above ground.

UNCLASSIFIED
D.C. NO. 9049

W

Calm 1.2%

E

S

06

Station: Knoxville, Tennessee

Date: Aug. 1948 - Feb. 1950

Data: Wind Rose for Gradient Wind
with rain.

30°
330°   20°
340°   10°
350°   0   350°
10   340°
20   330°
30°

40°
320°   320°
40°

50°
310°   310°
50°

60°
300°   300°
60°

70°
290°   290°
70°

80°
280°   280°
80°

90°
270°   W   E   270°
90°

100°
260°   260°
100°

110°
250°   250°
110°

120°
240°   240°
120°

130°
230°   230°
130°

140°
220°   220°
140°

45%
N
40%
35%
30%
25%
20%
15%
10%
5%

S

UNCLASSIFIED
DWG. NO.

07

Station: Knoxville, Tennessee

Date: Aug. 1948 - Feb. 1950

Data: Wind Rose for 10,000 Ft. with rain.

150°
210°   160°
200°   170°
190°   180°
180°   190°
170°   200°
160°   210°
150°

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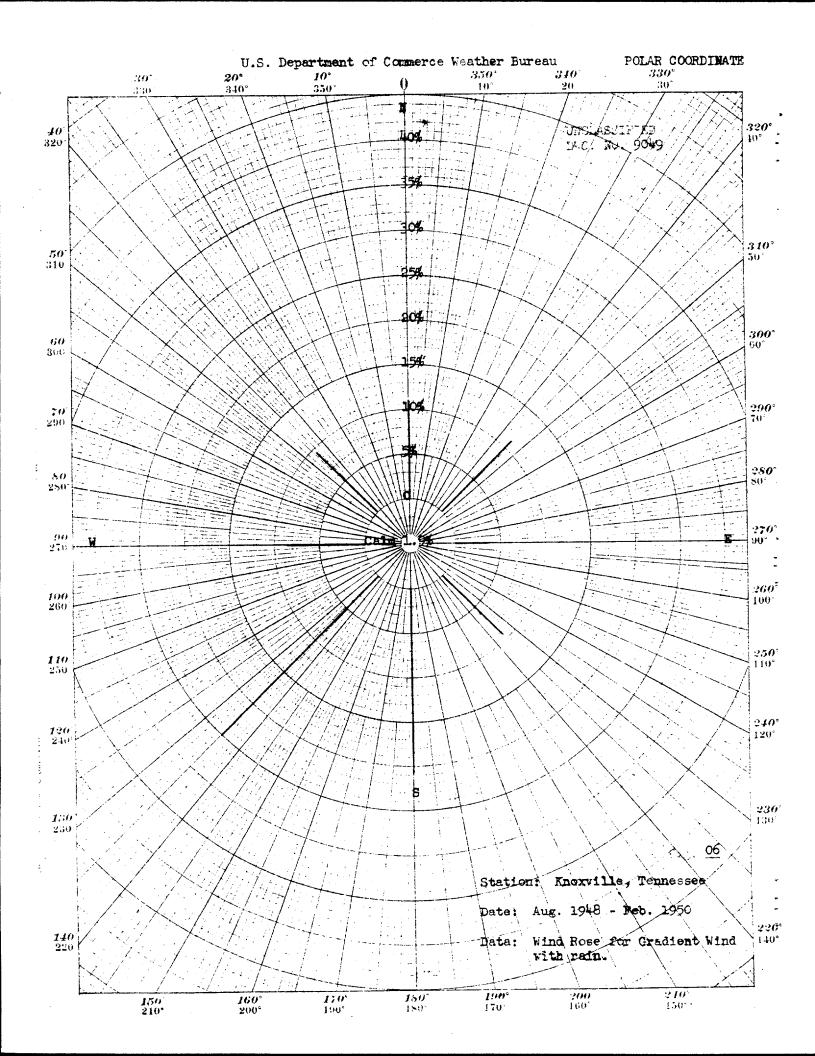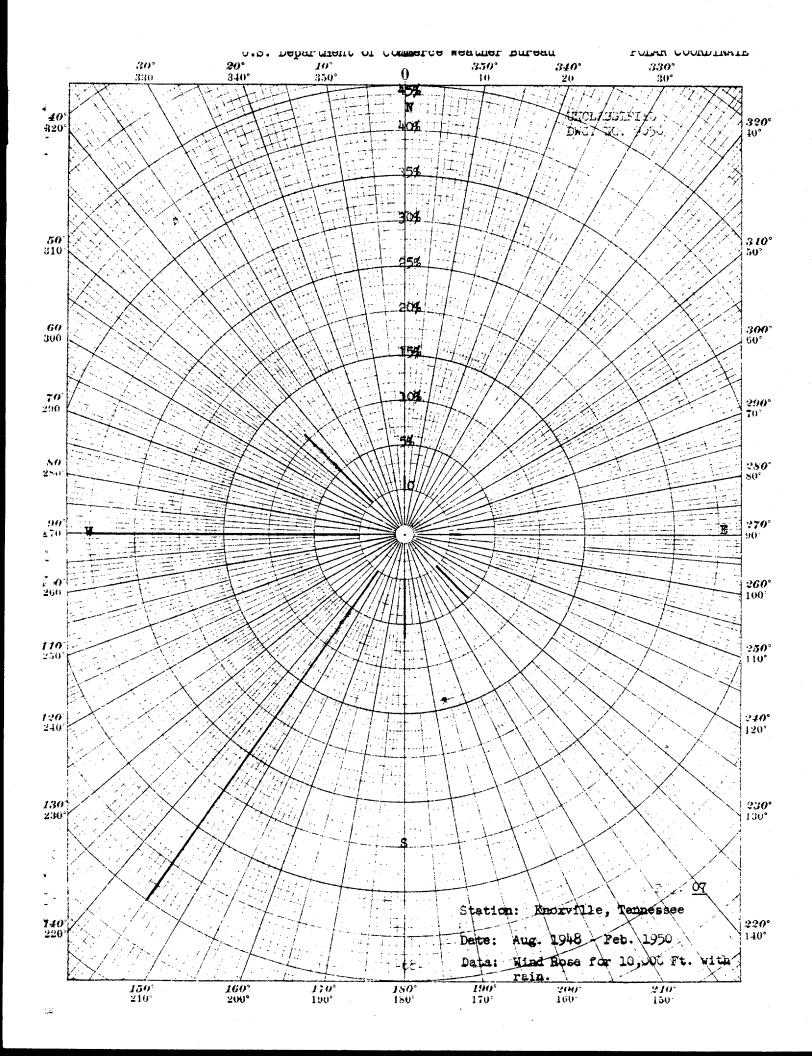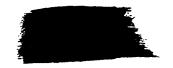
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UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY

UNITED STATES
TENNESSEE VALLEY AUTHORITY
MAPS AND SURVEYS DIVISION
(Windrock 129-SE)

TENNESSEE
BETHEL VALLEY QUADRANGLE
130-NE

Y-12

K-10

H.R.E. SITE

HALF MILE

ONE MILE

TWO MILES

Knoxville 25

1.3 MI. TO K-25

½ MILE RADIUS FROM K-25

4 MILE RADIUS FROM K-25

Fanny Knob

Gallaher Bend

Gallaher Bend

Melton Hill

Mountain View Sch

Gravel Hill

Jones Island

Clinch View

Woodlawn Ch

Macedonia Ch

Acme Sch

New Zion Ch

Wheat

George Jones Memorial Ch

Cumberland Ch

Mt. Vernon Ch.

Friendship Ch.

Douglas Chapel Cem

Pinhook Gap

Crystal Spring

East Fork Ch

Reservoir Elev. 741

Control by USC&GS, USGS, and TVA
Topography by Geological Survey from aerial
photographs by stereophotogrammetric methods
Field examination by Tennessee Valley Authority, 1941

Maps distributed by the Geological Survey

Approximate outline of Watts Bar Reservoir

APPROXIMATE MEAN DECLINATION, 1941

Scale 24000

Contour interval 20 feet
Datum is mean sealevel

1 2 Miles

Kilometers

Feet

Polyconic projection. 1927 North American datum
10,000 foot grid based on Tennessee rectangular
coordinate system
5,000 yard grid based on U. S. zone system, B

ROUTES USUALLY TRAVELED
HARD, IMPERVIOUS SURFACES
OTHER SURFACE IMPROVEMENTS
U. S. ROUTE   STATE ROUTE

BETHEL VALLEY, TENN
130-NE
Edition of 1941

N3552.5-W8415/7.5

Fig. 1
Dwg. 8584

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